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Calculation of Failure Importance Measures for Basic Events and Plant Systems in Nuclear Power Plants

Prepared by T. A. Wheeler, R. G. Spulak, Jr.

Sandia National Laboratories

Prepared for
U.S. Nuclear Regulatory
Commission

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ABSTRACT

This is a report on the project "Determination of Event Sequences Contributing to Overall Frequency of Core Melt." It is an analysis of thirteen existing probabilistic risk assessments (PRAs) covering fifteen power plants. The object of this project is to identify the plant systems and system components which contribute most to the overall core melt frequency and to some measure of risk. The results of this analysis are grouped into the two major types of commercial reactors, Boiling Water Reactors (BWRs), and Pressurized Water Reactors (PWRs). Within each type of reactor, plants are grouped by manufacturer. The results are first analyzed on a plant by plant basis. Then, the results are discussed for all plants within each plant manufacturer type to study any interrelationships or common trends which might exist among plants built by the same manufacturer. The results are then discussed among all PWRs and all BWRs.

The methodology developed for this analysis is based on dominant accident sequence cut sets comprised of system component level basic events. Only those accident sequences contributing five percent or more to the total core melt frequency or risk measure are considered as dominant in this analysis. These cut sets are taken directly from those PRAs which publish sequence cut sets. For those PRAs which did not include such logic models of the accident sequences, cut sets are taken from simplified fault trees developed from the sequence and system descriptions in the PRAs. This analysis is limited to the information supplied in the PRAs. No fault tree computer codes were utilized to generate cut sets for those PRAs which did not have cut sets. The frequency of initiating events and unavailabilities of basic events are based on data in the PRA documents. No allowance is made for plant specific or industry-wide changes which may have occurred since the various PRAs were published.

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EXECUTIVE SUMMARY

This is a report on the NRC project "Calculation of Failure Importance Measures for Basic Events and Plant Systems in Nuclear Power Plants." It is an analysis of thirteen existing probabilistic risk assessments (PRAs) covering fifteen nuclear power plants. The object of this project is to calculate failure importance measures for plant systems and system components with respect to three criteria or impact measures: 1) total frequency of core melt resulting in breach of the containment boundary, 2) total frequency of core melt, and 3) risk to the public due to core melt accidents. The analysis is performed on the two major types of commercial reactors, Boiling Water Reactors (BWRs), and Pressurized Water Reactors (PWRs). The importance calculations are first analyzed on a plant-by-plant basis. Then, the results are compared between plants within each type of plant manufacturer to study possible trends. The methodology of this study employs the Vesely-Fussell importance measure as the measure of a system's or system component's contribution to the various impact measures. The analysis is conducted at two levels, the front line and support system level, and the basic event or system component level. At the system level, the importance measure yields a conditional probability that, given a core melt accident occurs, some component or components of a particular system were involved in the accident failure sequence. At the basic event level, the importance measure has two quantitative interpretations. The first is, just as with the system level measures, a conditional probability that, given a core melt accident sequence occurs, a particular component is involved in the sequence of failures allowing the accident to occur. However, unlike the system level importance measure, the basic event level importance measure can be used to calculate the change in an impact measure (e.g. core melt frequency) due to a change in a particular component's unavailability. Thus, by using the basic event importance measures, the impact on core melt accident frequencies of improved component reliability can be assessed.

The Vesely-Fussell importance measure is calculated using Boolean equation logic models of the dominant accident sequences for each PRA. For those PRAs which did not publish Boolean equations, Boolean equations were derived from simplified fault trees constructed for this analysis from sequence and event descriptions in the PRA. No fault tree computer codes were utilized to generate sequence Boolean equations. This analysis used only those models published in the PRAs. No allowance was made for plant or industry-wide changes which may have occurred since the various PRAs were published.

The results of the importance calculations indicate that there is no strong trend of common systems or basic events which dominate core melt frequency or plant risk, either among different plant manufacturer types or within manufacturer types. Plant specific design can have significant impact as to which systems and events are found to be significant to core melt frequency or risk at a particular plant. In addition, differences in PRA methodology may result in different results between plants. For example, only four PRAs consider external events in analyzing accident sequences. In all four studies, external events were found to be very important. Yet even between these studies, the dominant external events vary depending on plant design and site features. Modeling differences in containment response, human error, recovery of component failures, and accident sequences can also yield different results between plants.

I. INTRODUCTION

A. Overview

The project "Calculation of Failure Importance Measures for Basic Events and Plant Systems in Nuclear Power Plants" is an analysis of thirteen existing probabilistic risk assessments (PRAs) covering fifteen nuclear power plants. These studies and plants are listed in Table 1. The object of this project was to calculate failure importance measures for plant systems and system components with respect to three criteria; total plant core melt frequency, total plant core melt frequency with release of radioactive material into the environment, and total plant risk due to core melt accident sequences. The analysis was performed on the two major types of commercial reactions, Boiling Water Reactors (BWR), and Pressurized Water Reactors (PWRs). Within each type of reactor, plants are grouped by manufacturer. The importance calculations are first analyzed on a plant-by-plant basis. Then, the results of all plants of the same manufacturer are compared to study any interrelationships or common trends which might exist among plants designed by the same manufacturer. The importance calculations are then discussed among all types of plant manufacturers to analyze any trends in important failures common to all plant types.

The methodology developed for this analysis is rigorously developed and explained in the following chapter, but an overview is given here. The importance measures are calculated from dominant core melt accident sequences published in each PRA. The cut sets for these sequences are taken directly from the PRA for those studies which published accident sequence cut sets. For those PRAs which did not publish such accident sequence logic models, cut sets are taken from simplified fault trees developed for this analysis from the sequence and system descriptions in the PRAs. The analysis is limited to information supplied in the PRAs. No fault tree computer codes were utilized to generate cut sets for those PRAs which did not publish cut sets. The frequency of initiating events and basic event unavailabilities are based on data in the PRA documents. No allowance is made for plant specific or industry-wide changes which may have occurred since the various PRAs were published.

The failure importance measures are calculated within the context of three criteria or impact measures. This enables a perspective to be gained as to how a system's or system component's importance to plant safety may vary with different value bases. Criterion 1 is with respect to the frequency of core melt sequences resulting in a release of radioactivity to the environment. Criterion 2 is with respect to total core melt frequency, and yields a measure of component and system contribution to the frequency of core melt, regardless of the consequences of a core melt

Table 1

Probabilistic Risk Assessments Analyzed

Plant Manufacturer	Plant	Study
Westinghouse	Sequoyah	Reactor Safety Study Methodology Applications Program (RSSMAP)
	Surry	WASH-1400
	Zion	Zion Probabilistic Safety Study
	Indian Point 2	Indian Point Probabilistic Safety Study (IPPSS)
	Indian Point 3	IPPSS
Babcock and Wilcox	Crystal River	Interim Reliability Evaluation Program (IREP)
	ANO-1	IREP
	Oconee	RSSMAP
Combustion Engineering	Calvert Cliffs	RSSMAP
General Electric	Browns Ferry-1	IREP
	Grand Gulf	RSSMAP
	Peach Bottom	WASH-1400
	Limerick	Limerick PRA
	Big Rock Point	Big Rock Point PRA
	Millstone 1	IREP

sequence. Criterion 3 is with respect to risk to the general public from a core melt sequence. This study is restricted to the basic events and systems involved in the dominant core melt sequences, or those sequences in each PRA which contribute 5 percent or more to the total impact measure for each criterion. The weight factors for criteria 1 and 3 are explained in Chapter II.

The failure importance measure calculated for the basic event, or system component, level analysis is the Vesely-Fussell importance measure. This importance measure is developed and defined mathematically in Chapter II. However, when the Vesely-Fussell importance measure is actually calculated for a basic event, the calculation is merely the sum over all the dominant sequences of the values (frequencies) of all the cut sets containing that basic event. Then, this sum is divided by the core melt frequency. In the case of Criteria 1 and 3, all values are weighted by the appropriate weight factors. The sum of the values of all cut sets containing a certain basic event can be intuitively interpreted as that portion of the core melt frequency or risk measure which involves the specific event. Dividing this sum by the appropriate impact measure (core melt, release of radioactivity, or plant risk) normalizes the basic events' contribution to the total impact measure and yields a dimensionless ratio. This ratio, the Vesely-Fussell importance measure, has two useful quantitative interpretations. As shown in Chapter II, the Vesely-Fussell term gives the fractional change in total core melt frequency or risk due to a given fractional change in the unavailability of a basic event, or component failure. The other interpretation of the measure is the conditional probability that, given a core melt, release, or death due to a core melt accident sequence, a particular basic event was involved in the accident sequence. These interpretations allow one to see which plant components are most crucial to plant safety, and how much impact on plant safety improved component reliability can yield.

The system level importance measure does not have the same quantitative significance as does the basic event importance measure. The system level importances are calculated by "labeling" each cut set by the various systems whose components appear in the cut set. Then, the values for all the cut sets which have been "labeled" by a particular system are summed, then divided by the total impact measure, just as with basic events. However, the system level measure cannot be used to show the fractional change in core melt frequency or risk due to a fractional change in system unavailability. This is because the concept of a single system unavailability is not valid. System success or failure is highly dependent upon success criteria for various accident sequences and on partial failures within a system. A more in-depth discussion of the difficulties in defining system unavailabilities is contained in Chapter II. The system level measure does provide a conditional probability that, given a core melt, release, or death due to a core melt accident, some component or components within the system, and thus the system itself was involved in the failure sequence of the accident.

B. Summary

The results of the various plants indicate that there is no strong trend of similar systems dominating core melt frequency or plant risk across or within manufacturer types. In the four studies which considered external events; Big Rock Point (BRP), Zion, and Indian Point 2 and 3; the external events tend to be significant to core melt frequency, and more significant than plant systems with respect to risk and containment failure. However, the actual external events which are dominant seem to be sensitive to site specific features. The studies which considered external events also used significantly different containment response models than the other PRAs. The Zion and Indian Point models greatly reduce the importance of plant systems to risk compared to the other PRA containment response models.

Operator errors are significant in ten of the fourteen PRAs analyzed. However, the type of operator errors and the level of importance of operator errors vary significantly from plant to plant. Furthermore, it is difficult to achieve an accurate measure of operator importance due to the nature in which unavailabilities developed from the PRAs which did not publish sequence cut sets contain significant human error contributions as well as hardware faults. But the manner in which the PRAs often describe system failures does not permit the human error contribution to be separated from the hardware faults. Even PRAs with published sequence cut sets such as Calvert Cliffs RSSMAP and Crystal River IREP often model human errors into the basic event unavailability.

Another discrepancy between PRAs which impacts the importance calculations is the manner in which recovery actions are modeled. Some PRAs model recovery actions as an explicit basic event. Other PRAs simply multiply a recovery factor with the value of an entire cut set. So the importance of recovery actions cannot always be calculated for PRAs. In general, PRA treatment of recovery models are weak, if recovery is discussed at all. This adds to the difficulty of trying to determine the importance of recovery actions.

The presentation of sequence models in the various PRAs presented significant difficulties for the application of the importance methodology for this analysis. Only six of the PRAs published sequence cut sets derived from sequence Boolean equations. These studies are Grand Gulf RSSMAP, Calvert Cliffs RSSMAP, Millstone IREP, Oconee RSSMAP, Crystal River IREP, and ANO-1 IREP. These studies are very compatible with the importance methodology of this analysis. The results for these plants present a very accurate calculation of the basic event and system importances for that portion of the core melt frequency modeled. However, as much as 30 percent of the cut sets for some of these PRAs were either not published in the PRA or not included in the dominant sequences. The importance results from the other nine PRAs are based on cut sets derived from highly simplified fault trees. For several of the PRAs, the basic events are virtually system and event tree level events. Importance calculations from such cut sets have a qualitative value for demonstrating which systems and events tend to be important for a certain plant. However, the actual quantitative values calculated for such generalized basic events should not be viewed as highly accurate importance measures.

II. METHODOLOGY

A. Methodology

To measure the contribution of systems and components of systems in risk assessments, we developed several importance measures and the methodology to calculate them. This section will describe this methodology and will discuss the uncertainties associated with the methodology. In addition, ideas for further research will be briefly presented.

Probabilistic Risk Assessments (PRAs) use fault tree techniques, usually linked with event trees for accident sequence delineation, to derive the frequencies of various accident sequences which lead to core melt. The consequences of the various scenarios of core melt are sometimes calculated. The risk from the plant is the expected product of accident frequency and consequences.

Several importance measures have been derived in the past for the importances of basic events of a fault tree to the top event. Lambert[1] presents a discussion of some importance measures in the context of coherent structure theory.[2] Such a theoretical development is necessary because fault trees are structures which represent Boolean algebra, and thus there is no guarantee that real algebra (and calculus, etc.) can be applied to fault trees or the equivalent Boolean equations.

The variables in a fault tree, including the top event, can have only the values 1 or 0. That is, either the component has failed or it has not. What we are interested in is the probability that the top event is 1 (or the probability that the top event occurs). To calculate this, we use failure data to represent the probabilities that the basic events are 1 (have failed). The fault tree is "solved"--The Boolean equation for the top event is written in disjunctive normal form.[3] This gives the minimal cut sets, or minimal combinations of basic events which must be 1 for the top event to be 1. The probabilities of these cut sets occurring, and thus the probability that the top event occurs, is calculated by using the Boolean expression for each cut set as a probability expression and substituting failure data for each Boolean variable. These failure variables can have any value between 0 and 1.

Fortunately, these probability expressions obey real algebra, and importance measures can be derived by formally manipulating the probability expressions (with real operations) which are totally equivalent to the importance measures derived with coherent structure theory.

Our approach in this study has been to derive importance measures using methods and notation which we believe can

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help the reader understand the significance and application of each importance measure. In reporting the results of our calculations, we have used the historical names of the importance measures (as commonly used in PRA literature). Table 2 lists the symbols we will use below to derive our importance measures.

The results of a PRA are usually presented by identifying the "dominant" cut sets of the "dominant" core melt sequences, and their values (including the probability of nonrecovery), and identifying the containment failure modes and probabilities, and release category assignments for the "dominant" core melt sequences. The consequences of releases in each release category are not always calculated. Our importance measures have been constructed so as to be easily calculable from the information usually given in published PRAs.

In general, there are three criteria which we can measure against to calculate the importances of systems and components. These are (1) total frequency of core melt resulting in breach of the containment boundary, (2) frequency of core melt, and (3) risk to the public. The criterion of interest depends upon the safety philosophy selected. That is, one can view all core melts of equal concern, whether any release is generated (Criterion 2), one can view all releases of equal concern independent of the magnitude and consequences (Criterion 1), or one can be concerned with the frequency of release and the consequences of the release, i.e., risk (Criterion 3). The analysis of risk performed in a PRA proceeds from calculating the core melt frequencies of various accident sequences by fault tree techniques, to assigning the sequences to release categories based on the type and amount of radioactive material released by each type of sequence, to calculating the consequences (the number of early and latent cancer induced deaths) due to releases in each release category. Since there is inevitably some uncertainty introduced at each stage of this process, the conservative approach would treat all core melts as equally bad. On the other hand, if one truly wishes to address public risk, the consequences of various accident types should be considered. This explains the range in safety philosophy and the three criteria used in this analysis.

Using the notation in Table 2, the total plant core melt frequency is

$$M_{\beta} = \sum_{j} (m_j)_{\beta} , \qquad (1)$$

Table 2

Notation Used for Derivation of Importance Measures

Symbol	Meaning
Q_i	Failure probability (or frequency) of component i
m_j	Frequency of core melt sequence j
α	Release category (subscript)
β	Nuclear Power Plant (subscript)
$P_{\alpha j}$	Probability that core melt sequence j has a release in release category α
C_α	Consequences of a release in release category α
I_α	Relative consequences of a release in release category α, normalized to the sum of release frequencies in all categories.
M_β	Total core melt frequency of Plant β
L_β	Total frequency of release of Plant β
R_β	Total Risk of Plant β
$S_{x,cs}$	Component failure probability, or product of failure probabilities, of system x in cut set cs.
$E_{x,cs}$	Set of Boolean variables which appear, either alone or as a product, in $S_{x,cs}$
$\frac{\Delta S_x}{S_x}$	Change by a given factor used to measure change in system x.

the frequency of release is

$$L_\beta = \sum_\alpha \sum_j (m_j P_{\alpha j})_\beta \quad , \tag{2}$$

and the total risk is

$$R_\beta = \sum_\alpha \sum_j (m_j P_{\alpha j} C_\alpha)_\beta \ . \tag{3}$$

These quantities are collectively referred to as "risk measures" for the power plant.

The sequence core melt frequency is given by the probability expression (cut sets) for the sequence:

$$m_j = \sum_{cs} \left(\prod_{i \epsilon c} Q_i \right)_j \ , \tag{4}$$

where the subscript cs denotes the set of all cut sets for sequence j, c denotes the set of variables in cut set cs, and the Q_i are the failure probabilities (or frequency) for the Boolean variables in each cut set. The Q_i include the sequence initiating event frequency. Note that this expression assumes the "small probability approximation" for the basic events.

The basic differential operator we use to operate on the probability expressions is the "logarithmic derivative":

$$\frac{dy}{d \ell n\ x} = x \frac{dy}{dx} \ , \tag{5}$$

which gives the change in y for a fractional, or percent, change in x.

The utility of this operator is that, for linear equations, it preserves the terms which contain the independent variable. This is important because PRAs usually only report cut set frequencies with nonrecovery included and do not give the values of the nonrecovery factors. (The cut sets are linear terms containing basic events as factors.) Thus it is much easier to calculate importance measures using the overall cut set frequencies instead of basic event values because there are some (implied) basic events (e.g., nonrecovery) for which additional calculations would be necessary to derive their probabilities.

Applying this operator to each of the risk measures, equations (1), (2), and (3) we have as importance measures for Boolean variable Q_k,

$$\frac{\partial M_\beta}{\partial \ell n \; Q_k} = \sum_j \left(\frac{\partial m_j}{\partial \ell n \; Q_k} \right)_\beta \quad , \qquad (6)$$

$$\frac{\partial L_\beta}{\partial \ell n \; Q_k} = \sum_\alpha \sum_j \left(P_{\alpha j} \frac{\partial m_j}{\partial \ell n \; Q_k} \right)_\beta \quad , \qquad (7)$$

and

$$\frac{\partial R_\beta}{\partial \ell n \; Q_k} = \sum_\alpha \sum_j \left(C_\alpha \; P_{\alpha j} \frac{\partial m_j}{\partial \ell n \; Q_k} \right)_\beta \quad . \qquad (8)$$

From Equation (4),

$$\frac{\partial m_j}{\partial \ell n \; Q_k} = \sum_{cs \mid k \in c} \left(\prod_{i \in c} Q_i \right)_j \; . \qquad (9)$$

That is, the logarithmic derivative of core melt sequence frequency j with respect to the probability value of Boolean variable k is simply the sum of the values of the cut sets which contain variable k. As discussed above, this allows us to use the cut set frequencies reported in a PRA without further complications. Equation (9) can be used with equations (6), (7), and (8) to calculate the values of $\partial M_\beta / \partial \; \ell n \; Q_k$, $\partial L_\beta / \partial \ell n \; Q_k$, and $\partial R_\beta / \partial \; \ell n \; Q_k$.

This importance measure, the logarithmic derivative of a probability expression, is the same as the "upgrading function" importance measure described by Lambert.[1] It can be used to calculate the change in the risk measure for a given change in the probability value of the Boolean variable:

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$$\Delta M_\beta = \frac{\Delta Q_k}{Q_k} \frac{\partial M_\beta}{\partial \ell n\ Q_k} \quad , \tag{10a}$$

$$\Delta L_\beta = \frac{\Delta Q_k}{Q_k} \frac{\partial L_\beta}{\partial \ell n\ Q_k} \quad , \tag{10b}$$

and

$$\Delta R_\beta = \frac{\Delta Q_k}{Q_k} \frac{\partial R_\beta}{\partial \ell n\ Q_k} \quad . \tag{10c}$$

These expressions are derived from Taylor's series expansions for ΔM_β, ΔL_β, and ΔR_β, and are exact because all higher order derivatives vanish. Since the change in each case is proportional to the importance measure multiplied by the fractional change in the variable ($\Delta Q_k/Q_k$), ranking the variables by the values of these importance measures will order them by the magnitude of their effect on the risk measure for a given factor or fractional (%) change. Note that since the importance measure is a partial derivative, it is possible to change only one basic event probability, Q_k, at a time and calculate the effect. Part C of this chapter will discuss how it would be possible to calculate the effect of changing two or more basic event probabilities at a time.

The other importance measure which we calculate is equivalent to the "Vesely-Fussell" importance measure.[1] This is calculated by dividing the "upgrading function" importance measure by the appropriate risk measure:

$$\frac{\partial \ell n\ M_\beta}{\partial \ell n\ Q_k} = \frac{1}{M_\beta} \frac{\partial M_\beta}{\partial \ell n\ Q_k}, \tag{11a}$$

$$\frac{\partial \ell n\ L_\beta}{\partial \ell n\ Q_k} = \frac{1}{L_\beta} \frac{\partial L_\beta}{\partial \ell n\ Q_k}, \tag{11b}$$

and

$$\frac{\partial \ell n\ R_\beta}{\partial \ell n\ Q_k} = \frac{1}{R_\beta} \frac{\partial R_\beta}{\partial \ell n\ Q_k}. \tag{11c}$$

Our interest in this importance measure lies in its interpretation. While the "upgrading function" is more easily used to calculate changes in the risk measures (Equations (10a), (10b), and (10c)), the "Vesely-Fussell" measure gives the probability that the basic event occurred given that the "top event" occurred. For example, $\partial \ell n M_\beta / \partial \ell n Q_k$ is the probability that component k failed (and contributed to the accident) given that the core has melted. Since the "upgrading function" is just divided by a constant to determine the "Vesely-Fussell" importance measure, the "Vesely-Fussell" measure does not contain different information. The only motive for using the "Vesely-Fussell" measure is ease of interpretation.

Table 3 gives a summary of the definitions, methods of calculation, and interpretation of the component importance measures. Note that in practice we do not use the actual consequences, C_α, of a release in each release category. This is because many PRAs have not performed consequence calculations and the actual consequences are site specific. While the actual consequences may be unknown, we have defined the relative consequences of a release in each release category based on consequence calculations for a composite northeast river valley site.[4] The relative consequences are

$$I_\alpha = \frac{C_\alpha}{\sum_\alpha C_\alpha} \quad . \tag{12}$$

These relative consequences may be less sensitive to the site than the absolute consequences.[5] It should be recognized that the importance measures we calculate for risk are, in effect, normalized to the consequences at this composite site. We have defined a normalized risk, $\bar{R}_\beta$,

$$\bar{R}_\beta = \sum_j \sum_\alpha \; (I_\alpha P_{\alpha j} \, m_j)_\beta \quad , \tag{13}$$

which replaces R_β in Equations (8), (10c), and (11c). Thus our "upgrading function" cannot be used to calculate actual changes in risk, but rather changes in the normalized risk measure. However, the "Vesely-Fussell" measures have the normalization divided out, so they can be interpreted as the probability of failure of the component given that offsite consequences (fatalities) have occurred.

Table 4 gives the relative consequences of WASH-1400 release categories as calculated with the CRAC code for the

Table 3

Component Importance Measures for Plant β

	Criterion (Risk Measure)	Definition of Sensitivity	Name	Operation	Interpretation
1.	Frequency of Release	$\frac{\partial L_{\beta}}{\partial \ell n\ Q_k}$	"Upgrading function"	Sum cut sets containing Q_k, weighted by probabilities of release in release categories in which sequences appear.	Used to calculate change in frequency of release when probability value of component k is changed.
		$\frac{\partial \ell n\ L_{\beta}}{\partial \ell n\ Q_k}$	"Vesely-Fussell"	Divide above by total frequency of release.	Probability that component k failed given a release occurs.
2.	Core Melt Frequency	$\frac{\partial M_{\beta}}{\partial \ell n\ Q_k}$	"Upgrading function"	Sum cut sets containing Q_k.	Used to calculate change in core melt frequency when probability value of component k is changed.
		$\frac{\partial \ell n\ M_{\beta}}{\partial \ell n\ Q_k}$	"Vesely-Fussell"	Divide above by total core melt frequency.	Probability that component k failed given core melt.
3.	Risk	$\frac{\partial \bar{R}_{\beta}}{\partial \ell n\ Q_k}$	"Upgrading function"	Sum cut sets containing Q_k, weighted by probabilities of release and relative consequences of categories in which sequences appear.	Used to calculate change in normalized risk when probability value of component k is changed.
		$\frac{\partial \ell n\ \bar{R}_{\beta}}{\partial \ell n\ Q_k}$	"Vesely-Fussell"	Divide above by total normalized risk.	Probability that component k failed given offsite fatalities.

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Table 4

Relative Consequences* of WASH-1400 Release Categories[4]

Relative Consequences

Release	PWR			BWR		
Category	Early	Latent	Total	Early	Latent	Total
1	0.930	0.610	0.680	0.88	0.50	0.510
2	0.066	0.180	0.150	0.12	0.32	0.320
3	0.004	0.140	0.110		0.17	0.160
4		0.050	0.037		0.01	0.009
5		0.016	0.012			
6		0.003	0.002			
7						

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*Consequences = fatalities

composite northeast river valley site.[4] Note that this table reflects early, latent cancer induced, and total fatalities. We have used the relative consequences based on total fatalities for the I_α.

Most PRAs have assigned accident sequences to WASH-1400 release categories. The one real exception, the Big Rock Point PRA, defined a different set of release categories and performed consequence calculations for those release categories. To calculate the importance measures for Big Rock Point, we defined I based on the consequences and categories in that PRA. Thus the Big Rock Point results for "Criterion 3" (risk) are not normalized to the site in Reference 4. In addition, our Limerick PRA analysis included correlating the Limerick containment failure modes with WASH-1400 release categories.

We have described the methodology used to calculate importance measures for the basic faults (or system components) contained in the cut sets reported in published

PRAs. We can also derive importance measures for systems, but interpretation of the meaning of system importance measures is more difficult. This is because a cut set for an accident sequence does not contain a failure probability for a system as a whole. Because different accidents have different system success requirements, and initiating events and support system failures may fail some or all of a system, the cut sets contain failures of components and groups of components within a given system. There is, in general, no overall system failure which can be defined which is appropriate to all accident sequences (or even all cut sets in a given sequence). That is, writing $\partial M_\beta / \partial \ln A_x$ for the logarithmic derivative of total core melt frequency of plant β with respect to the unavailability of system x, A_x, is meaningless because there is no well-defined variable, A_x.

However, we can define a system variable which is a function of the system component unavailabilities for which we can write a derivative. The importance measures so defined can be interpreted meaningfully. Because the definition of failures of a system depends not only on the system design but on the accident sequences and even cut sets which are considered, our measure of system performance is a function of both the system of interest and the risk model (PRA) of the plant, i.e. cut sets. Let $S_{x,cs}$ be the elements of a set

$$\left\{ S_{x,cs} \right\} \equiv \begin{array}{l}\text{The set of the system x component failure}\\ \text{probabilities or combinations (products)}\\ \text{of component failure probabilities which}\\ \text{appear in each of the cut sets, cs.}\end{array} \qquad (14)$$

Define

$$\frac{\partial}{\partial \ln S_x} \equiv \sum_{cs} \frac{\partial}{\partial \ln S_{x,cs}} , \qquad (15)$$

and

$$\left[\frac{\Delta S_x}{S_x}\right] \quad \begin{array}{l}\text{change of all the elements of}\\ \left\{S_{x,cs}\right\} \text{ by the same factor,}\\ \Delta S_{x,cs}/S_{x,cs}.\end{array} \qquad (16)$$

Note that we do not define a value for S_x; it is a dummy variable which measures changes in the system. Also,

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$S_{x,cs}$ may contain many identical elements since the same system component or combinations of components may appear in many sequence cut sets. Then,

$$\frac{\partial m_j}{\partial \ell n \; S_x} = \sum_{cs} \frac{\partial m_j}{\partial \ell n \; S_{x,cs}} \quad j = \sum_{cs|\; E_{x,cs} \epsilon c} \left(\prod_{i \epsilon c} Q_i \right)_j ; \qquad (17)$$

where $E_{x,cs}$ is the set of Q_i which appear in $S_{x,cs}$. ($E_{x,cs}$ is either a single Q_i or the set of Q_i which appear as a product.)

Equation (17) can be used with Equations (6), (7), and (8) to define $\partial M_\beta/\partial \ell n \; S_x$, $\partial L_\beta/\partial \ell n \; S_x$, and $\partial \bar{R}_\beta/\partial \ell n$ x. These are the "upgrading functions" for systems. Then the changes in the risk measures are:

$$\Delta M_\beta = \left[\frac{\Delta S_x}{S_x}\right] \frac{\partial M_\beta}{\partial \ell n \; S_x} , \qquad (18a)$$

$$\Delta L_\beta = \left[\frac{\Delta S_x}{S_x}\right] \frac{\partial L_\beta}{\partial \ell n \; S_x} , \qquad (18b)$$

and

$$\Delta \bar{R}_\beta = \left[\frac{\Delta S_x}{S_x}\right] \frac{\partial \bar{R}_\beta}{\partial \ell n \; S_x} . \qquad (18c)$$

These equations can be used to calculate changes in overall core melt frequency, frequency of release, and risk for changes in the system defined by $[\Delta S_x/S_x]$.

The "Vesely-Fussell" measures for systems can also be calculated

$$\frac{\partial \ell n \; M_\beta}{\partial \ell n \; S_x} = \frac{1}{M_\beta} \frac{\partial M_\beta}{\partial \ell n \; S_x} , \qquad (19a)$$

$$\frac{\partial \ell n \; L_\beta}{\partial \ell n \; S_x} = \frac{1}{L_\beta} \frac{\partial L_\beta}{\partial \ell n \; S_x} , \qquad (19b)$$

and

$$\frac{\partial \ell n\ \bar{R}_\beta}{\partial \ell n\ S_x} = \frac{1}{\bar{R}_\beta} \frac{\partial \bar{R}_\beta}{\partial \ell n\ S_x} \quad . \tag{19c}$$

The "Vesely-Fussell" measures are easily interpreted as the probability that any component or set of components in the given system failed given that the "top event" occurred. For example, $\partial \ell n\ M_\beta/\partial \ell n\ S_x$ is the probability that some combination of components in system x failed, and contributed to the accident, given that the core has melted. Thus the "Vesely-Fussell" measure is intuitively meaningful and useful for ranking systems as to their importance.

The use of the system-level "upgrading function" is not nearly so straightforward. A problem arises because it is difficult to define a change in the system which corresponds to a change in failure probability. As discussed above, this is because there is, in general, no system failure probability variable which is consistent between all sequences or even all cut sets of a given sequence. We have defined a system variable, $S_{x,cs}$, and a change $[\Delta S_x/S_x]$, which lead to system importance measures which are formally consistent with those defined for components. However, use of the change in system performance, $[\Delta S_x/S_x]$, to calculate changes in the risk measures may be difficult to interpret.

The change $[\Delta S_x/S_x]$ is defined so that all the elements of $\{S_{x,cs}\}$ are changed by the same factor, $\Delta S_{x,cs}/S_{x,cs}$. This means that values of all the cut sets containing any component or combination of components of the system are changed by the same factor. While simple enough in principle, physically changing a system to produce the same factor change in failure probability for these components or combinations of components may be impossible. For example, suppose the following components and combinations of components of a system appear in accident sequence cut sets: a, a*b, and b. (There is no Boolean reduction since each cut set also contains failures of other components of other systems.) Thus to calculate a change in a risk measure (say ΔM_β), the three $S_{x,cs}$ (Q_a, $Q_a{*}Q_b$, and Q_b) must be changed by the same factor. That is,

$$\frac{\Delta Q_a}{Q_a} = \frac{\Delta Q_b}{Q_b} = \frac{\Delta(Q_a{*}Q_b)}{Q_a{*}Q_b} \quad . \tag{20}$$

However,

$$\frac{\Delta(Q_a{*}Q_b)}{Q_a{*}Q_b} = \frac{\Delta Q_a{*}Q_b + Q_a{*}\Delta Q_b + \Delta Q_a{*}\Delta Q_b}{Q_a\ Q_b}$$

$$= \frac{\Delta Q_a}{Q_a} + \frac{\Delta Q_b}{Q_b} + \frac{\Delta Q_a}{Q_a}\,\frac{\Delta Q_b}{Q_b}, \qquad (21)$$

which is never equal to $\Delta Q_a/Q_a$ or $\Delta Q_b/Q_b$.

The fundamental reason that the "upgrading function" for systems is difficult to interpret is that the cut sets are not linear with respect to components of the system.

Once again, the system-level "Vesely-Fussell" measure can be used to rank system importances in a meaningful way, but calculating changes in risk measures due to changes in system-level data, or other inputs, is difficult to interpret. Table 5 gives a summary of the definitions, methods of calculation, and interpretation of the system importance measures.

B. Uncertainties

The product of this study, i.e., the importances of components and systems as measured against the three importance criteria, is subject to four areas of uncertainty. Three of these uncertainties arise from the limitations of the PRAs reviewed while the other arises from our methodology.

Table 6 gives a description of each area of uncertainty. This section will discuss these uncertainties and quantify their impacts on our results.

The first two uncertainties affect the numerical values derived in the PRA for component unavailabilities, sequence frequencies, and the overall core melt frequency and risk. These uncertainties therefore affect our results directly. That is, since we derive the sensitivity of core melt frequency, release frequency, and risk to the values of component unavailabilities, our results are uncertain to the extent that the original data and overall evaluation of risk from the plant are uncertain.

The other two uncertainties affect the interpretation of the results of our analysis. The PRA is a model of unreliability (as measured by core melt frequency or risk) of the nuclear power plant. Since we operate on only a portion of this model, we can judge the sensitivity of only that portion of the model to system unavailability.

Table 5

System Importance Measures for Plant B

Criterion (Risk Measure)	Name	Operation	Interpretation
1. Frequency of Release	"Upgrading function"	Sum cut sets containing any component(s) of system x, weighted by probabilities of release in release categories in which sequences appear.	Used to calculate change in frequency of release when failure probabilities of portions of system x which appear in cut sets are changed by some factor.
	"Vesely-Fussell"	Divide above by total frequency of release.	Probability that some portion of system x failed given a release occurs.
2. Core Melt Frequency	"Upgrading function"	Sum cut sets containing any component(s) of system x.	Used to calculate change in frequency of release when failure probabilities of portions of system x which appear in cut sets are changed by some factor.
	"Vesely-Fussell"	Divide above by total core melt frequency.	Probability that some portion of system x failed given core melt.
3. Risk	"Upgrading function"	Sum cut sets containing any component(s) of system x, weighted by probabilities of release and relative consequences of categories in which sequences appear.	Used to calculate change in normalized risk when failure probabilities of portions of system x which appear in cut sets are changed by some factor.
	"Vesely-Fussell"	Divide above by total normalized risk.	Probability that some portion of system x failed given offsite fatalities.

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Table 6

Areas of Uncertainty

Source of Uncertainty	Description of Impact
1. Data base used in PRA for basic faults.	Uncertainty associated with numerical values of cut set and sequence frequencies.
2. Methodology of PRA.	Uncertainty of completeness of characterization of risk from plant.
3. Reporting of PRA.	Uncertainty associated with nature (systems) and value of sequences and cut sets not reported as dominant. Also, lack of information on various events and assumptions.
4. Truncation in this methodology.	Uncertainty from considering only sequences contributing more than 5 percent.

Examining each source of uncertainty (Table 6) in turn, we first consider the data base used in the PRA. In general, the PRAs we considered in this study use component failure data developed by the Reactor Safety Study (RSS, WASH-1400). Table 7 gives the sources of data for representative PRAs. The commonality of the WASH-1400 data base has two implications. First, we can compare importances derived from different PRAs and be confident that the differences arise from the differences in plant design and operation, not the data. Second, the numerical uncertainties in our results will generally reflect the numerical uncertainties in the WASH-1400 data base. In general, the assessed failure rates in WASH-1400 for system components had 5 and 95 percent bounds on a lognormal distribution which were factors of 1/3 and 3 times the mean, respectively. Human actions had larger error bounds, usually 1/10 and 10 times the mean. The results of the RSS PRA were not sensitive to the exact interval the bounds were chosen to represent, e.g., 85 percent bound versus 95 percent. This is because these bounds are on the tails of the distribution.

The combinations of the failure distributions through the fault tree analysis result in cut set frequencies with uncertainties which depend on the square root of the number of basic events in the cut sets. Since our results are derived from the same data and methods, our results will generally have this uncertainty, also. This means that, for example, component and system importances which differ by factors of 2 or less should be judged as roughly equivalent, while differences greater than factors of 2 are actually significant.

The second source of uncertainty is from incompleteness of the knowledge of the actual unreliability of the plant as calculated by the PRA. That is, PRAs generally do not consider fires, floods, aircraft crashes, and other "external" events as contributors to risk. Also, assumptions are made in methodology which may be untrue (e.g., the initiating event for an accident sequence does not disable some safety systems). And, there are uncertainties in knowledge of phenomena, eg, the probability of a steam explosion in the reactor vessel. If the actual risk were to be dominated by some factor not adequately addressed in the PRA, we would greatly overestimate the sensitivity of risk to the factors which were considered. These issues have been discussed in some detail, especially in the Reactor Safety Study. Some believe that the PRA methods do characterize the dominating portions of risk from nuclear power plants. This is not certain, of course, but it would not be meaningful to speculate on the quantitative impact of this uncertainty on our analysis. It is, however, important to view our results in the context of the scope of PRA.

Table 7

Data Sources for PRAs

PRA	Data Base
RSS - PWR (Surry) - BWR (Peach Bottom)	RSS
RSSMAP - Oconee - Sequoyah - Grand Gulf - Calvert Cliffs	RSS with the exception of a few plant specific failure rates (e.g., PCS availability).
IREP - Crystal River - Browns Ferry - Arkansas Nuclear One - Calvert Cliffs - Millstone	RSS with the exception of a few failure rates modified by data from LERs.
Limerick	RSS, General Electric, and LER generated data, (General agreement between three sources.)

The next uncertainty (as listed in Table 6) is due to the limited state of our knowledge of the models of risk developed by the PRA. That is, even if the PRA discovered all risks from the plant and modeled them correctly, the fact that not all accident sequences, and cut set sets for accident sequences, are reported limits the interpretation of our sensitivity analysis on that model of risk. This is because we must assume in our ignorance that improvements in reliability of components contributing to the dominant sequences (which are reported) either do or do not affect the risks associated with the nondominant sequences (which are not reported). The concern is that we perform a sensitivity analysis on only a portion (albeit dominant portion) of the risk model developed by the PRA. This is due to both inadequacy (for our purposes) of reporting of the PRA and our truncation of the sequences we consider at 5 percent of the value of the criterion being addressed. We were requested by the USNRC to consider only those accident sequences which contribute more than 5 percent to the overall plant core melt frequency or risk. This is the last source of uncertainty in Table 6. The model we operate on is a portion of the set of cut sets for the dominant accident sequences (and the release category assignments of those sequences). Using the core melt frequency, M_β, for illustrative purposes, we calculate the importance of a system, k, to that portion of M_β we have knowledge of, M_β:

$$\partial M_\beta' / \partial \ln Q_k,$$

where Q_k is the probability variable for component k and appears in cut sets for sequences contributing to core melt. What we are trying to discover is how different this importance is from the actual importance of system k:

$$\partial M_\beta / \partial \ln Q_k$$

To do so, we need to know how changing Q_k affects that portion of M_β we do not have knowledge of, M_β'' (so $M_\beta = M_\beta' + M_\beta''$). There are three reasonable assumptions we can make about the contribution of Q_k to M_β'':

1. Q_k appears in no cut sets for M_β'' ,

2. Q_k appears in all cut sets for M_β'' ,

3. Q_k appears in the cut sets which contribute the same fraction of core melt frequency to M_β'' as to M_β' .

The first assumes we make no error in neglecting M''_β, the second assumes we make the largest possible error in neglecting M''_β, and the third assumes the only reasonable alternative between the two extremes. The importance measures in the three cases are then

$$\frac{\partial M_\beta}{\partial \ell n\ Q_k} = \frac{\partial M'_\beta}{\partial \ell n\ Q_k} , \tag{22a}$$

$$\frac{\partial M_\beta}{\partial \ell n\ Q_k} = \frac{\partial M'_\beta}{\partial \ell n\ Q_k} + fM_\beta \tag{22b}$$

and

$$\frac{\partial M_\beta}{\partial \ell n\ Q_k} = \left(\frac{1}{1-f}\right) \frac{\partial M'_\beta}{\partial \ell n\ Q_k} , \tag{22c}$$

for assumptions 1, 2, and 3, respectively. These equations give the importance of system k to the actual core melt frequency, M_β, for the three assumptions where f is the fraction of core melt frequency for which we do not know the cut sets and, once again, $\partial M_\beta / \partial \ell n\ Q_k$ is the importance we calculate by operating on those cut sets we do have. These three equations can also be used for the frequency of release and risk criteria by substituting L_β or $\bar{R}_\beta$ for M_β and using the fraction of L_β or $\bar{R}_\beta$ not operated on for f.

The three importance measures, Equations (22a), (22b), and (22c), were calculated for each component or system considered and were used to judge the degree of confidence with which to view our results for each PRA.

Another aspect of incomplete reporting of PRAs is that we had to develop a methodology consistent with what was habitually reported in the published documents. There are some important limitations inherent in our approach. First, the importance measures for importances of components and systems to core melt frequency ("Criterion 2") may be somewhat uncertain because the sequence accident frequencies were simply added, both in the original PRAs and our analysis, to derive the plant core melt frequency. If the sequences were to be combined with AND logic, some Boolean reduction might occur. This is because the sequences were defined to distinguish between accidents with different phenomenology and consequences so that some of the core melt sequences are merely other core melt sequences with mitigating system failures added. The core melt would occur with or without the mitigating system failures, so summing these sequence frequencies overestimates the total core melt frequency. This uncertainty does not apply to the importances to risk ("Criterion 3") because proper account is taken of the differences in consequences between sequences.

Another limitation is that many PRAs allow operator recovery actions, but do not explicitly report which cut sets are recoverable and what the non-recovery probabilities are. Since failure to recover can be viewed as an operator error, and we could not always distinguish these actions in our analysis, the importances of operator action may be underestimated.

C. Further Development

One limitation of the importance measures we have derived is that we can calculate changes in the risk measures from changing only one basic probability value at a time. The methodology can be extended to allow calculation of the changes in risk measures from changing any number of basic probability values.

The value of the top event of a fault tree can be thought of as a dependent function of n independent variables (the basic event probabilities, Q_i). The value of the top event can be expressed as a Taylor expansion around the value as calculated by the PRA.

The general form of a Taylor expansion in two variables is:[6]

$$f(x,y) = \sum_{m=0}^{\infty} \sum_{n=0}^{\infty} \frac{1}{m!n!} \frac{\partial^{m+n}}{\partial_x m \, \partial_y n} f(a,b)(x-a)^m (y-b)^n \quad , \qquad (23)$$

where f is the dependent variable, x and y are the independent variables, and x = a, y = b is the point expanded around. In our notation, we have, for core melt frequency,

$$\Delta M_\beta = \frac{\Delta Q_k}{Q_k} \frac{\partial M_\beta}{\partial \ell n Q_k} + \frac{\Delta Q_\ell}{Q_\ell} \frac{\partial M_\beta}{\partial \ell n Q_\ell} + \frac{\Delta Q_k}{Q_k} \frac{\Delta Q_\ell}{Q_\ell} \frac{\partial^2 M_\beta}{\partial \ell n Q_k \, \partial \ell n Q_\ell} \quad , \qquad (24)$$

with similar expressions for frequency of release and risk by substituting L_β or R_β for M_β, where components k and ℓ are being changed simultaneously. Note that, since the cut sets are linear in Q_k and Q_ℓ, all higher order terms vanish. In addition, if Q_k and Q_ℓ are both less than zero, then the last term in Equation (24) makes the effect on ΔM_β less from changing both simultaneously than would be expected from simply adding the independent effects.

Finally,

$$\frac{\partial^2 M_\beta}{\partial \ell n Q_k \partial \ell n Q_\ell} = \left(\sum_{cs|k,\ell \epsilon c} \left(\prod_{i \epsilon c} Q_i \right) \right)_\beta . \qquad (25)$$

That is, the second derivative is the sum of the frequencies of the cut sets which contain both Q_k and Q_ℓ. Similar expressions can be generated for changing three, four, or n basic probabilities simultaneously.

III. IMPORTANCE OF SYSTEMS AND SYSTEM COMPONENTS

To evaluate the importance of systems and system components to the three criteria of this study, the results have been grouped by the two major types of light water reactors, PWRs and BWRs. Within this grouping, plants are grouped by the plant vendors. Within each vendor type, each plant is treated individually, then results between plants of the same vendor are discussed. Results among all PWRs and BWRs are discussed after each plant vendor type has been treated. For discussion in this chapter we use the Vesely-Fussell importance measures. In addition, "importance" refers to the importance as calculated with equation (22a), "worst case importance" corresponds to equation (22b), and "proportional importance" corresponds to equation (22c).

Certain aspects of the results discussed in this section may cause some confusion, so they are described here to better facilitate understanding of the analysis. In all but three PRA studies (Zion, Indian Point 2, and Indian Point 3), the assumption was made that given a core melt sequence, there will be some sort of failure of the containment. Thus, with the exception of the three studies given above, the results for Criterion 1 and Criterion 2 are exactly equivalent. Only the Zion and Indian Point studies contend that certain core melt sequences will be fully contained. For all the PRA studies, it is entirely possible for sequences which are dominant in Criteria 1 and 2 to not be dominant in Criterion 3, and visa versa. How a sequence ranks in Criterion 3 is dependent upon the distribution of the sequence's release category probabilities across the various WASH-1400 release categories. Thus, a sequence which is very significant to core melt frequency may be of little significance to plant risk if its probabilities of release in the high risk release categories is very low. It is also possible for the sum of all the release category probabilities for a single sequence to be greater than 1.0. An example of this is on Table 8: Sequence S_1D for Sequoyah has a probability of 1.0 for release category 5, and a probability of 0.01 for category 1. These probabilities are taken directly from the various PRAs, and show a conservative tendency to deal with the uncertainty of release category probabilities. This conservatism is incorporated into the analysis of this report.

In comparing the importance of a front line or support system to the importance of the system's components (or basic events of that system's failure) one should realize that the two importances, that of the system and that of the component, are both with respect to total core melt frequency or plant risk, dependent on the Criterion in question. The importance of the component is not its

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importance to the unavailability of the system. All importance calculations are based on accident sequence cut sets. These cut sets are derived from fault tree models which describe the logic of accident sequences. The cut sets, and therefore, the importance calculations, can only be related to the top event (i.e., core melt), and not to the unavailability of front line and support systems.

It is entirely possible that, for a certain system, the sum of the importance of all its components is greater than the importance of the system itself. This can occur when components are shared by more than one system. The significance of the system level results is to afford a generalized view of how plants might fail. It is the component level importance which must be used to determine the impact of plant changes on the core melt frequency or risk.

Whenever possible, initiating events from non-RSSMAP PRAs were grouped into RSSMAP initiating event terms for the system level analysis. Often, there is a very close correlation between RSSMAP initiating events and those of other PRAs. However, in those cases where a correlation could not be made, the original initiating event was either used by itself or grouped into a PRA specific initiating event group.

A. Importance of Systems and Components-PWR Results

1. Westinghouse Plants:

Sequoyah
Surry (WASH-1400 PWR)
Zion
Indian Point 2
Indian Point 3

a. Sequoyah

Analysis on the Sequoyah 1 power plant was performed using the Sequoyah Reactor Safety Study Methodology Applications Program (RSSMAP) PRA. This PRA was prepared for the US Nuclear Regulatory Commission (NRC) by Sandia National Laboratories. This study was the first RSSMAP Study performed and is not as detailed in its analysis as are the other RSSMAP studies. Cut sets were not calculated for the Sequoyah PRA. In this report simplified sequence fault trees were developed using the event trees in the PRA and the event descriptions to determine the basic sequence logic. Then, the system descriptions in the appendices were used to complete the fault trees and define the basic events. The fault trees and cut sets developed for the Sequoyah analysis are shown in Appendix B.1. All sequences are taken directly from the RSSMAP Study. The

Boolean terms defined for the simplified fault trees are listed on Table 10. The system level terms are listed on Table 9.

1. Criteria 1 and 2, Total Core Melt Frequency

The dominant sequences for Criteria 1 and 2 are shown in Table 8, along with the sequence frequencies and category release probabilities. The importances of the front line and support systems are listed on Table 11, and graphically displayed in Figure 1. The fraction of core melt frequency for which cut sets are unknown in this analysis is 0.12. The importance calculations are thus based on 88 percent of the core melt frequency. Considering the uncertainties of data used in PRAs and the limited information provided in many PRAs, 88 percent is reasonably close to the full core melt frequency.

Figure 1 shows seven system level terms with significant importance to core melt frequency. Not surprisingly, the LOCA initiating events S_2 and S_1 are among the most important terms, both at the system level and component level. Since every single cut set must have an initiating event and all but one dominant sequence is an S_1 or S_2 sequence, these initiating events are expected to have a large importance.

The most significant system level terms are operator error or some other human error involving plant personnel (OPERROR), and High Pressure Injection System (HPIS). High Pressure Recirculation System (HPRS), Low Pressure Recirculation System (LPRS), and Interfacing System LOCAs (V) follow with relatively low importances of .1 or less. The contribution of the Containment Sprays Recirculation System (CSRS) is insignificant.

Inspection of Figure 2 shows that the three most important basic events are the three human error terms, OPVRC, OPVSUC, and CMSD, all with an importance of .15. The hardware failures, CCHW, LPHW, and HPHW, are slightly less important, with the single event sequence V about half as important as the human errors.

The basic events, OPVRC, OPVSUC, HPHW, and LPHW, appear in the four dominant sequences involving failure of recirculation cooling to the primary system, S_1H, S_2H, S_1HF, and S_2HF. The slight differences in their importances appear to be due to the small differences in their values as basic events, as listed in Table 10. This follows from their appearance in the same sequences, but in independent cut sets. The event, CMSD, appears in only two sequences, S_1HF and S_2HF, which involve failure to supply water to the sump. But CMSD appears in two event cut sets, which are relatively high in value with respect to many of the cut sets containing the other terms. Thus, its importance is of the same magnitude as the other dominant terms. Likewise, CCHW appears in two event cut sets in the two sequences involving failure of injection systems, S_1D and S_2D.

The human error terms involve improper operator actions during switchover to recirculation (OPVSUC, OPVRS), or improper alignment of drains between upper and lower containment after refueling (CMSD). OPVSUC and OPVRC are explained in further detail in the Sequoyah PRA B.10, as common mode failures. CMSD is described as a common mode failure in the Sequoyah PRA B.8.

The HPIS hardware failure, CCHW, is primarily due to failures in the charging pump lines, and SIHW is due to safety injection pump failures. These are explained in more detail in the Sequoyah PRA A9. The HPRS hardware failure, HPHW, is dominated by insufficient ventilation air flow to the changing pumps during recirculation. This event is more fully explained in the Sequoyah A10. Hardware failures in the LPRS (LPHW) are dominated by a double failure. Both suction lines of the LPRS fail to provide sufficient flow. This failure is included under double failures in Sequoyah B.8.

Event V is a single event interfacing systems LOCA created by failure of check valves separating the Reactor Cooling System (RCS) from the LPIS. This event is described in Appendix A.5 of the Sequoyah PRA.

The result of the Sequoyah analysis indicate that human error dominates importance to core melt frequency at the system level. yet at the basic event level, hardware failure terms have importances relatively close in magnitude to the individual human error terms. This fact, combined with the independent nature of the basic events, suggests that all of the events discussed above should be considered as potential areas of interest in any attempt to reduce core melt frequency.

2. Criterion 3, Total Plant Risk

Only four Sequoyah accident sequences are dominant to plant risk; S_2H, S_1HF, S_2HF, and V. The PWR risk factors used for weighting release categories with respect to risk show release categories 1, 2, and 3 to clearly dominate risk. Sequences S_1D, S_2D, and S_1H have little little probability of release in these categories (Table 8), so it is not surprising that these sequences become nondominant for plant risk.

Table 8

Sequoyah-1

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Westinghouse	4	Ice Condenser	1148	RSSMAP

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures		
		1	2	3
1	S_1H	.17	.17	-
2	S_2H	.30	.30	.37
3	S_1HF	.05	.05	.06
4	S_2HF	.10	.10	.16
5	S_1D	.06	.06	-
6	S_2D	.11	.11	-
7	V	.08	.08	.14

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study PWR Release Categories							Sequence Frequency $(R\text{-}yr)^{-1}$
		1	2	3	4	5	6	7	
1	S_1H	.01				1.0			9.4E-6
2	S_2H			1.0					1.7E-5
3	S_1HF			1.0					2.9E-6
4	S_2HF		1.0						5.5E-6
5	S_1D					1.0			3.4E-6
6	S_2D				1.0				6.3E-6
7	V		1.0						4.6E-6

Total Plant Core Melt Frequency 5.6E-5

Table 9

Sequoyah
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
AFWS	Auxiliary Feed Water System	APAHW APATM APBHW APBTM APTHW APTTM IV
CSRS	Containment Spray Recirculation System	CSRS
EACPOWER	Emergency A.C. Power System	B3
ERCW	Essential Room Cooling Water System	ERCW
HPIS	High Pressure Injection System	CCHW SIHW
HPRS	High Pressure Recirculation System	HPHW
LPRS	Low Pressure Recirculation System	LPHW LPTM
OPERROR	Operator Error	CMSD OPURC OPUSUC
PCS	Power Conversion System	PCSTZ PCST23
S1	Small LOCA (2"≤D<6)	S1
S2	Small LOCA (D<2")	S2
T1	Loss of offsite power transient	T1
T2	Transient with automatic trip and main feedwater interruption	T2
T3	Transient with automatic trip but no feedwater interruption	T3
V	Low Pressure Injection System Check valve rupture	V

Table 10

Sequoyah RSSMAP
Boolean Terms From Sequence Fault Trees

Boolean Term		Value
APAHW	Auxiliary feedwater system (AFWS) pump hardware failure in Loop A.	2.2E-3
APBHW	AFWS pump hardware failure in Loop B.	2.2E-3
APATM	Unavailability of AFWS pump A due to test and maintenance.	1.1E-2
APBTM	Unavailability of AFWS pump B due to test and maintenance.	1.1E-2
APTHW	AFWS hardware failure in turbine pump.	5.3E-3
APTTM	Unavailability of AFWS turbine pump due to test and maintenance.	1.1E-2
B3	Failure of emergency AC power systems.	1.0E-3
CCHW	Hardware failure in charging lines or pumps of the High Pressure Injection System (HPIS).	2.8E-3
CMSD	Common mode contributor to failure of the Emergency Coolant Recirculation System (ECRS) and the Containment Spray Recirculation System (CSRS) caused by failure to open drains between upper and lower containment compartments. These drains are closed during refueling operations.	3.0E-3
CSRS	Failure of the Containment Spray Recirculation System (CSRS).	3.3E-3
ERCW	Essential Raw Cooling Water System failure to provide cooling water to AFWS pumps.	1.0E-2
HPHW	Hardware failure in the High Pressure Recirculation System (HPRS).	2.0E-3
IV	Failure of a single isolation valve between condensate storage tanks and AFWS pumps.	3.1E-3

Table 10 (continued)

Sequoyah RSSMAP
Boolean Terms From Sequence Fault Trees

Boolean Term		Value
LPHW	Hardware failure in Low Pressure Recirculation System (LPRS).	1.4E-3
LPTM	Failure of LPRS due to test and maintenance of one suction line and failure of the other line	1.6E-4
OPVRC	Operator failure to realign LPRS and HPRS for recirculation mode.	3.0E-3
OPVSUC	Operator failure to open valves in suction lines to LPRS discharge.	3.0E-3
PCST1	Failure of Power Conversion System (PCS) during transient with loss of offsite power.	2.0E-1
PCST23	Failure of PCS during a transient without loss of offsite power.	1.0E-2
S1	Small LOCA (2 in. D 6 in.).	9.8E-4/R-yr
S2	Small LOCA (D 2 in.).	1.8E-3/R-yr
SIHW	Hardware failure in Safety Injection lines or pumps of the HPIS.	7.0E-4
T1	Transient with loss of offsite power.	.2/R-yr
T2	Transient with automatic trip and main feedwater interruption.	3/R-yr
T3	Transient with automatic trip but no feedwater interruption.	4/R-yr
V	Low Pressure Injection System Check valve rupture.	4.6E-6/R-yr

Table 11

Sequoyah-1 RSSMAP

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
S_2	.52	.63	.59
OPERROR	.44	.57	.51
S_1	.28	.40	.32
HPIS	.17	.30	.20
HPRS	.10	.22	.11
V	.08	.20	.09
LPRS	.08	.20	.09
CSRS	.002	.12	.002

Table 12

Sequoyah-1 RSSMAP

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
S_2	S_2	.52	.64	.59
OPERROR	OPVRC	.15	.27	.17
	OPVSUC	.15	.27	.17
	CMSD	.15	.27	.17
S_1	S_1	.28	.40	.32
HPIS	CCHW	.14	.26	.16
	SIHW	.04	.18	.04
HPRS	HPHW	.10	.22	.14
V	V	.08	.20	.09
LPRS	LPHW	.07	.19	.08
	LPTM	.008	.13	.009
CSRS	CSRS	.002	.12	.002

Table 13

Sequoyah-1 RSSMAP

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
S_2	.53	.80	.73
OPERROR	.46	.72	.62
V	.14	.40	.19
HPRS	.08	.34	.11
S_1	.06	.33	.09
LPRS	.06	.33	.08
CSRS	.002	.27	.003

Table 14

Sequoyah-1 RSSMAP

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk Frequency: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
S_2	S_2	.53	.80	.73
OPERROR	CSMD	.24	.48	.30
	OPVRC	.12	.38	.16
	OPVSUC	.12	.38	.15
V	V	.14	.40	.19
HPRS	HPHW	.08	.34	.11
S_1	S_1	.06	.33	.09
LPRS	LPHW	.05	.32	.07
	LPTM	.006	.27	.009
CSRS	CSRS	.002	.26	.003

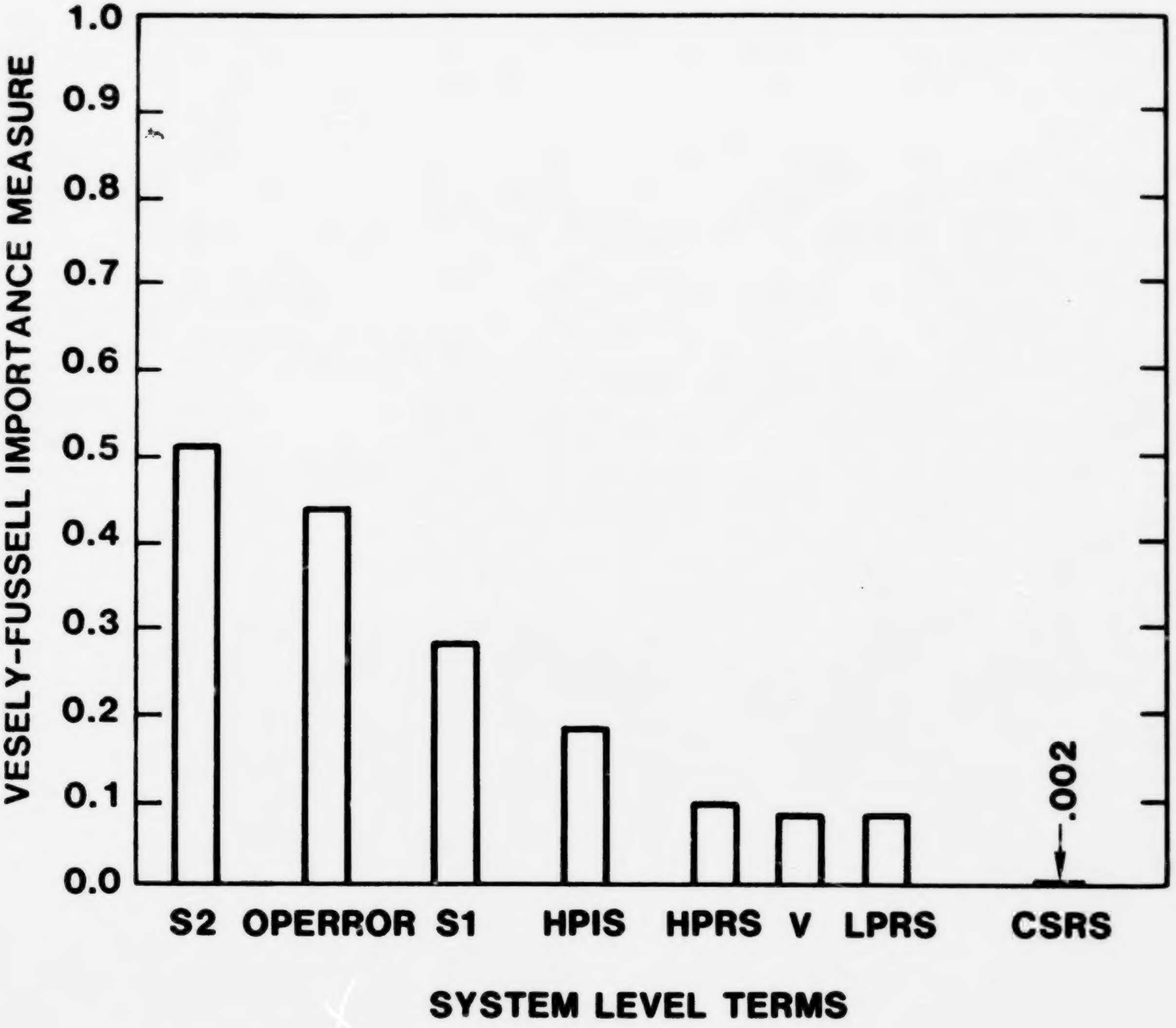

Figure 1. Sequoyah-1, Front Line and Support Systems Importance to Total Core Melt: Criteria 1 and 2

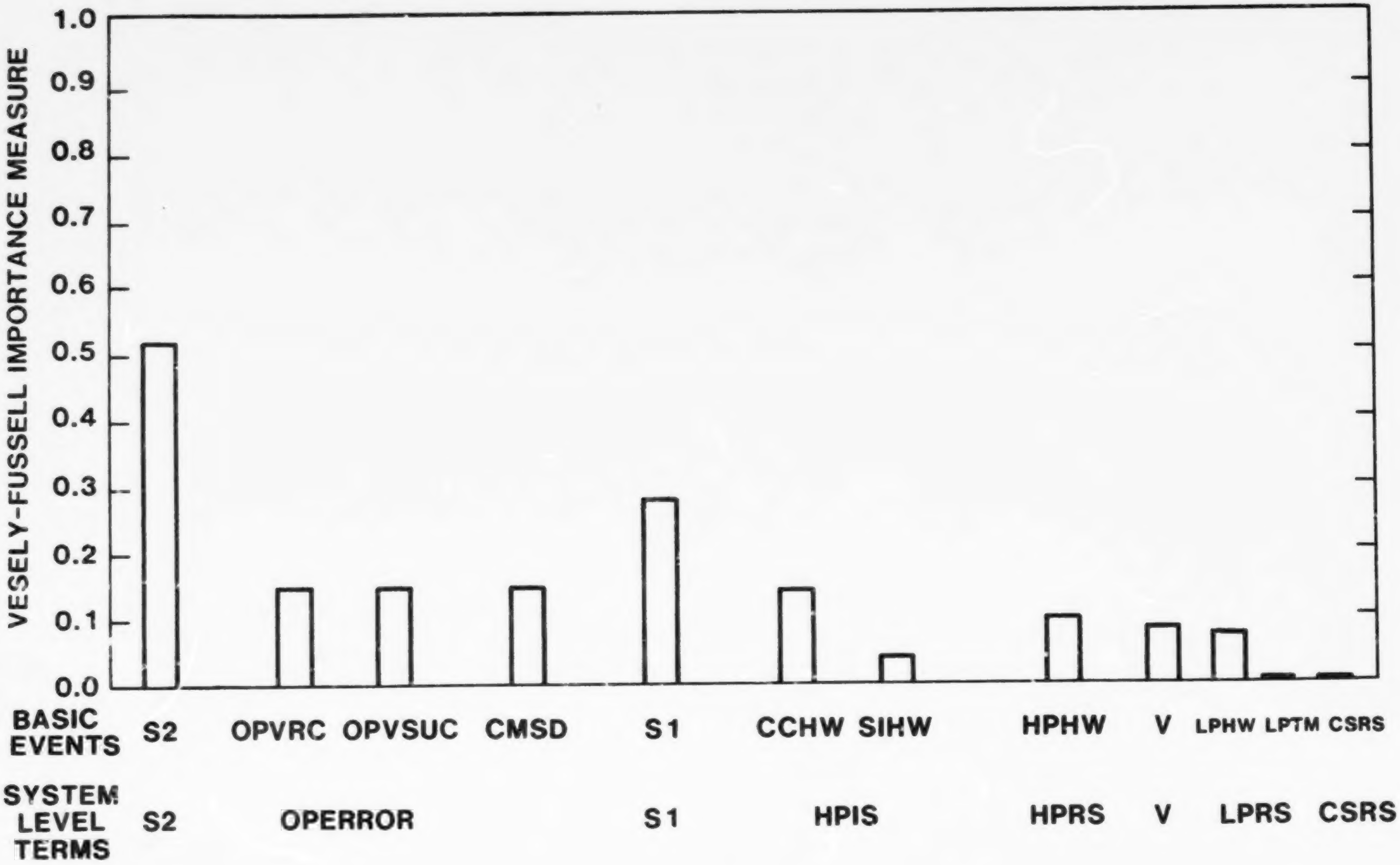

Figure 2. Sequoyah-1, Importance of Basic Events, Grouped by Front Line and Support Systems, to Total Core Melt Frequency: Criteria 1 and 2

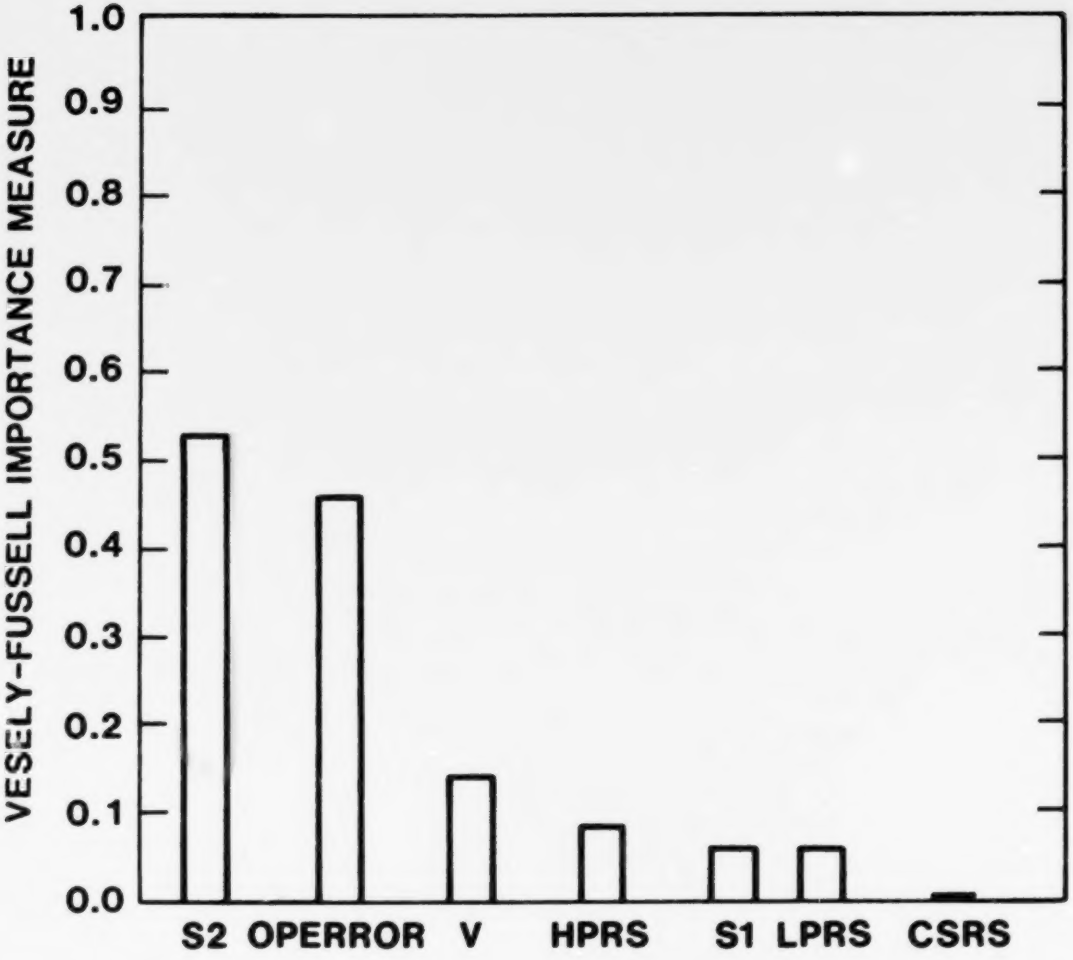

Figure 3. Sequoyah-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

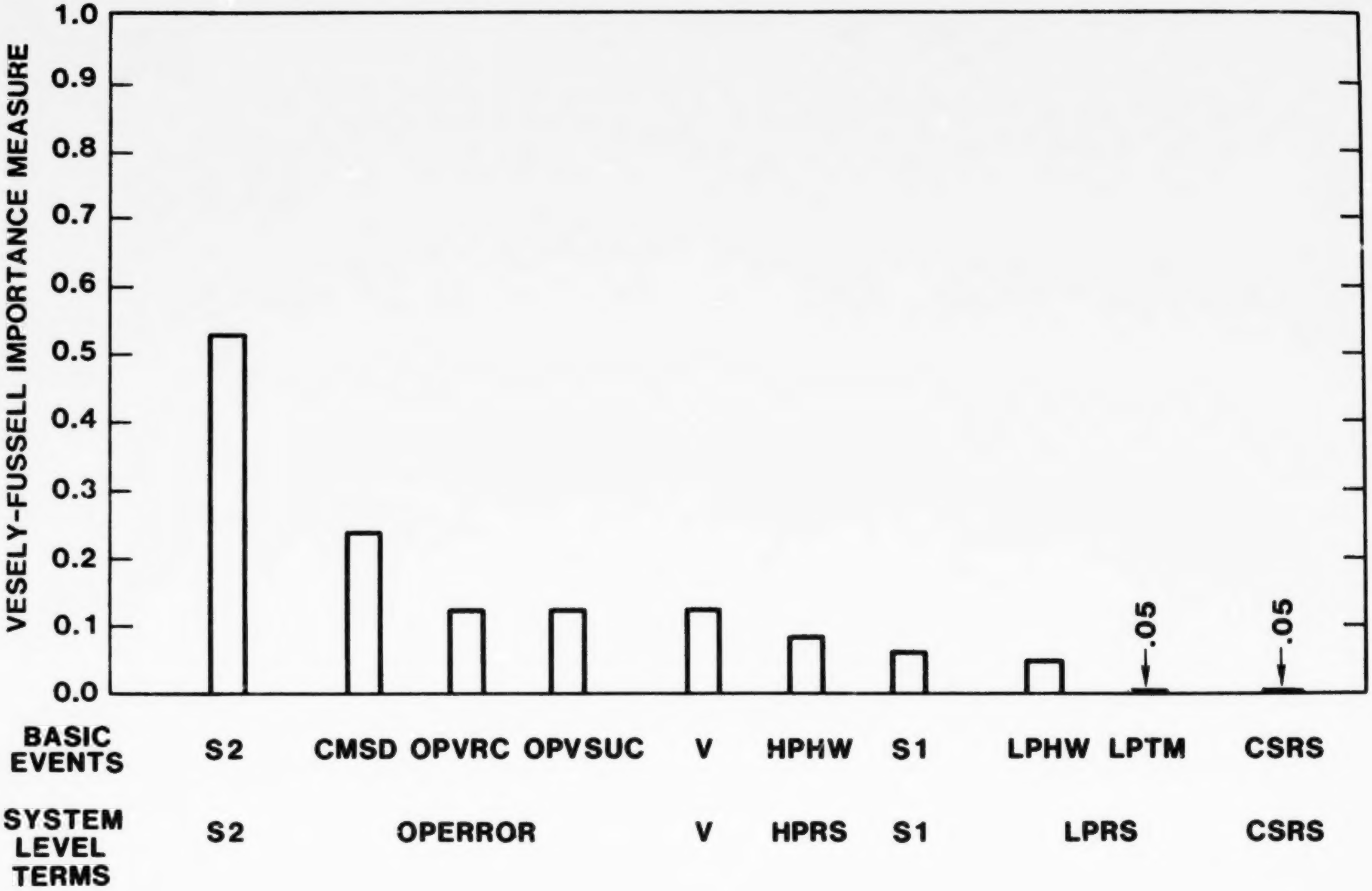

Figure 4. Sequoyah-1, Importance of Basic Events Grouped by Front Line Systems and Support Systems to Total Plant Risk: Criterion 3

Table 13 contains the dominant system level terms for Criterion 3, and Table 14 contains the importances of the basic events for each system level term. Since three of the dominant sequences are recirculation failures (S_2H, S_1HF, S_2HF), it is not surprising that many of the systems dominant to risk are also dominant to core melt frequency.

Figures 3 and 4 show that system level and component level terms rank similarly with respect to plant risk as they do with respect to core melt frequency. However, there are three significant differences. HPIS and its components are not important to risk. Sequence (and basic event) V is more important for Criterion 3 than Criteria 1 and 2, both relative to other events and absolutely. Human error term CMSD is clearly the single most dominant basic event. Event V is more important because its release category probability (Table 8) is 1.0 for category 2. Sequence S_2HF also has a high release probability in category 2. Event CMSD occurs in the most dominant cut set of sequence S_2HF. This, combined with the high risk release probability of the sequence, increases CMSD's importance over the other events.

b. Surry, (WASH-1400 PWR)

The analysis of the Surry nuclear power plant was done using the Reactor Safety Study (WASH-1400). The study was done by the NRC. Neither fault trees nor cut set information for the dominant accident sequences were published in WASH-1400. For this analysis, simplified fault trees were developed for the dominant sequences. Cut sets were derived using these simplified trees. The fault trees were developed by using the event trees in the PRA to define the events of each sequence. The basic events were then developed from the descriptions of systems in Appendix II of WASH-1400. These fault trees are shown in Appendix B.2. The fault trees for Surry are only developed down to a very general level, and there is almost no difference between the system level terms and the systems' components, as illustrated in Table 16 and Figure 6. The system level terms are defined in Table 16, and the basic events are defined in Table 17.

1. Criteria 1 and 2, Total Core Melt Frequency

The sequences dominant to core melt frequency are listed in Table 15. Approximately 6 percent of the core melt frequency is not represented by the cut sets of these sequences. This indicates that the simplified fault trees developed for Surry, though very general in detail, model 94 percent of the core melt frequency.

The system level results are shown in Table 18, and the basic event importances are in Table 19. These results are illustrated in Figures 5 and 6. In order to represent WASH-1400 transient terms as RSSMAP transient terms at the system level, RSSMAP transients T_2 and T_3 have more than one corresponding WASH-1400 transient. The transient terms RLOP and RLOP/actually are not transients, but merely the fractions of TFW transients involving loss of offsite power or not. Thus, although these terms do appear in cut sets, their importance values are of no significance.

The only other dominant systems level terms with more than one basic event are RECOVERY, AFWS, and LPRS. RECOVERY is used as a catch-all system level term throughout this analysis. Basic events of RECOVERY are any plant personnel actions which fail to restore systems and components.

Figures 5 and 6 show, not surprisingly, that initiating events are very dominant. Six basic events contribute more than 10 percent to core melt frequency, with High Pressure Injection (HPIS) and Auxiliary Feedwater (AFWST) failures the most dominant.

The cut sets for the dominant sequences (Appendix B2) show that the high pressure and low pressure cooling failures in both the injection and recirculation modes (HPIS, HPRS, HPLPRS) are independent of all the other dominant events. Similarly, event V, the interfacing systems LOCA, is independent of all other events. Auxiliary Feedwater failures and failure of AC power, both onsite (EMAC) and offsite (NROP1), appear in common cut sets.

The HPIS failures are dominated by single and double failures. The AFWS is dominated by test and maintenance failures in the turbine train. The HPRS and LPRS failures are dominated by human errors in aligning systems properly for recirculation. Emergency AC power failures are dominated by the failure of two diesel generators. These failures are described in detail in WASH-1400 Appendix II.

2. Criterion 3: Total Plant Risk

Table 15 shows only three accident sequences which are dominant to plant risk. Only TMLB', V, and S_2C have significant release probabilities in high consequence release categories (1, 2, and 3). The system level and basic event importances to risk are listed in Tables 20 and 21, and are illustrated in Figures 7 and 8.

Emergency core cooling systems are not important to risk, as sequences involving these systems do not dominate risk. The importances of the AFWS, RECOVERY, and AC power failures (EMAC, NROP1, NROP3, and NREMAC) are equal. This is because sequence TMLB' has only one cut set.

Table 15

Surry (WASH-1400 PWR)

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Westinghouse	3	Dry, Subatmospheric	775	RSS (WASH-1400)

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures		
		1	2	3
1	TMLB'	.08	.08	.32
2	V	.09	.09	.45
3	S_2D	.21	.21	-
4	S_1H	.07	.07	-
5	S_1D	.07	.07	-
6	S_2H	.23	.23	-
7	TML	.14	.14	-
8	TKQ	.07	.07	-
9	S_2C	-	-	.17

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study PWR Release Categories							Sequence Frequency
		1	2	3	4	5	6	7	$(\text{R-yr})^{-1}$
1	TMLB'	.009	.82				.18		3.3E-6
2	V		1.00						4.0E-6
3	S_2D			.01		.002		.99	9.1E-6
4	S_1H			.01		.002		1.00	3.0E-6
5	S_1D			.01		.002		1.00	3.0E-6
6	S_2H			.02		.002		.98	9.9E-6
7	TML			.02		.0001		1.00	6.1E-6
8	TKQ			.01		.0001		1.00	3.0E-6
9	S_2C	.008		1.00					2.0E-6

Total Plant Core Melt Frequency 4.4E-5

Table 16

Surry
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
A	LOCA (D 6")	A
ACC	Accumulator System	ACC
AFWS	Auxiliary Feedwater System	AFWST AFWS
CHRS	Containment Heat Removal System	CHRS
CSIS	Containment Spray Injection System	CSIS
CSRS	Containment Spray Recirculation System	S2
EACPOWER	Emergency AC Power System	EMAC
HPIS	High Pressure Injection System	HPIS
HPRS	High Pressure Recirculation System	HPRS
LOPNRE	Non-recovery of offsite power, early	NROP1
LOPNRL	Non-recovery of offsite power, late	NROP3
LPIS	Low Pressure Injection System	LPIS
LPRS	Low Pressure Recirculation System	HLPRS LPRS
RECOVERY	Non-recovery of AC power/feedwater System	NREMAC NRMFW1
RPS	Reactor Protection System	RPS
RV	Reactor Coolant Relief Valve	RV
S1	LOCA (2" D 6")	S1
S2	LOCA (1/2" D 2")	S2
T1	Transient Caused by Loss of Offsite Power	RLOP TFW TLOP

Table 16 (Cont.)

Surry
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
T2	Transients Involving Loss of Main Feedwater, but Not Loss of Offsite Power	RLOP/ TFW
T3	Any Transient Other Than T_1 or T_2	T
V	Low Pressure Injection Check Valve	V

Table 17

Surry
Boolean Terms from Sequence Fault Trees

Boolean Term	Description	Value
A	LOCA D 6"	1 E-4/R-yr
ACC	Failure of Accumulators	8.3E-4
AFWS	Failure of the Auxiliary Feedwater System Given No Loss of Offsite Power	3.5E-5
AFWST	Failure of the Auxiliary Feedwater Turbine Train	1.5E-2
CHRS	Failure of the Containment Heat Removal System	8.5E-5
CSIS	Failure of the Containment Spray Injection System	2.4E-3
CSRS	Failure of the Containment Spray Recirculation System	1.0E-4
EMAC	Failure of Emergency (onsite) AC Power	1.0E-2
HLPRS	Failure of that Portion of the Low Pressure Recirculation System (LPRS) Which Feeds the High Pressure Recirculation System (HPRS)	4.9E-3
HPIS	Failure of the High Pressure Injection	8.6E-3
HPRS	Failure of the High Pressure Recirculation System	5.0E-3
LPIS	Failure of the Low Pressure Injection System	4.7E-3
LPRS	Failure of the Low Pressure Recirculation System	1.3E-2
NROP1	Non-recovery of Offsite Power in 1/2-1 Hour	0.2
NROP3	Non-recovery of Offsite Power in 3 Hours	0.5
NREMAC	Non-recovery of Emergency AC Power in 3 Hours	1.0

Table 17 (Cont.)

Boolean Term	Description	Value
NRMFW1	Non-recovery of the Main Feedwater System in 1/2-1 Hour	1.0E-2
RLOP	Fraction of Transients Involving Loss of Main Feedwater which are Loss of Offsite Power	6.7E-2
RLOP/	Fraction of Transients Involving Loss of Main Feedwater which are Not Loss of Offsite Power. RLOP/= 1 - RLOP	0.93
RPS	Failure of the Reactor Protection System	3.4E-5
RV	Reactor Coolant System Relief Valve Fails to Reseat	1.0E-2
S1	LOCA 2" D 6"	1E/.R-yr
T	Any Transient	10/R-yr
TFW	Transient with Loss of Main Feedwater	3/R-yr
TLOP	Transient from Loss of Offsite Power	0.2/R-yr
V	Low Pressure Injection System Check Valve Failure	4.0E-6

Table 18

Surry

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
S_2	.42	.48	.45
HPIS	.25	.32	.27
AFWS	.23	.29	.24
T_1	.20	.27	.22
EACPOWER	.20	.27	.22
LOPNRE	.20	.27	.22
HPRS	.15	.21	.16
LPRS	.15	.21	.15
S_1	.13	.19	.14
V	.09	.15	.10
RECOVERY	.09	.15	.10
T_3	.07	.13	.07
RV	.07	.13	.07
RPS	.07	.13	.07
LOPNRL	.07	.13	.07
T_2	.02	.08	.02
ACC	.006	.07	.006

Table 19

Surry

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
S_2	S_2	.42	.48	.45
HPIS	HPIS	.25	.32	.27
AFWS	AFWST	.21	.26	.22
	AFWS	.02	.03	.02
T_1	TFW	.16	.22	.17
	RLOP	.14	.20	.15
	TLOP	.07	.13	.07
EACPOWER	EMAC	.20	.27	.22
LOPNRE	NROP1	.20	.27	.22
HPRS	HPRS	.15	.21	.16
LPRS	HLPRS	.15	.21	.15
S_1	S_1	.13	.19	.14
V	V	.09	.15	.10
RECOVERY	NREMAC	.07	.13	.07
	NRMPW1	.02	.08	.02
T_3	T	.07	.13	.07
RV	RV	.07	.13	.07
RPS	RPS	.07	.13	.07
LOPNRL	NROP3	.07	.13	.07
T_2	TFW	.16	.22	.17
	RLOP1	.02	.08	.02
ACC	ACC	.006	.07	.006

Table 20

Surry

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
V	.45	.53	.49
AFWS	.29	.37	.32
T_1	.29	.37	.32
EACPOWER	.29	.37	.32
LOPNRE	.29	.37	.32
LOPRNRL	.29	.37	.32
RECOVERY	.29	.37	.32
CSIS	.17	.26	.19
S_2	.17	.26	.19

Table 21

Surry

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
V	V	.45	.53	.49
AFWS	AFWST	.29	.37	.32
T_1	TLOP	.29	.37	.32
EACPOWER	EMAC	.29	.37	.32
LOPNRE	NROP1	.29	.37	.32
LOPNRL	NROP3	.29	.37	.32
RECOVERY	NREMAC	.29	.37	.32
CSIS	CSIS	.17	.26	.19
S_2	S_2	.17	.26	.19

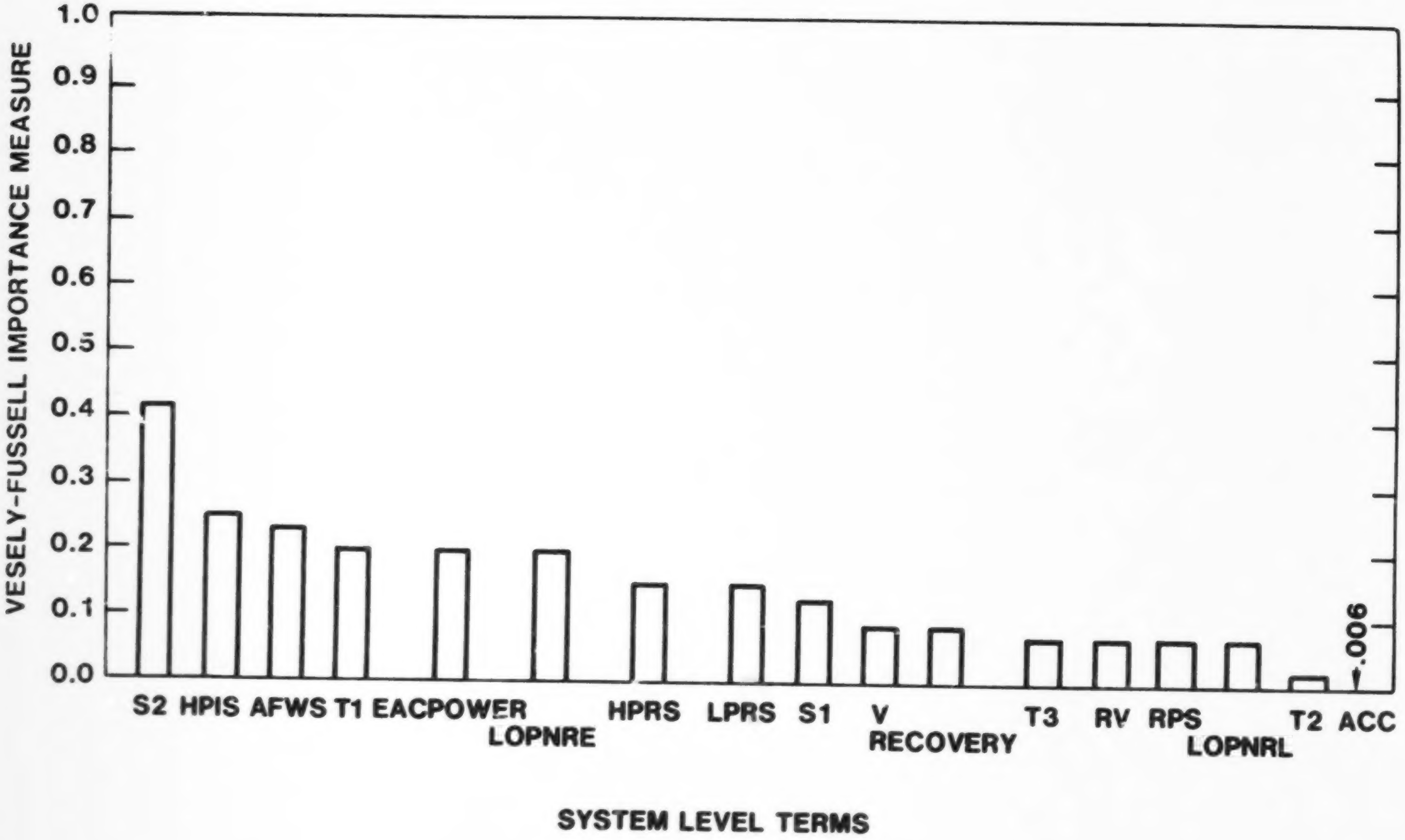

Figure 5. Surry, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

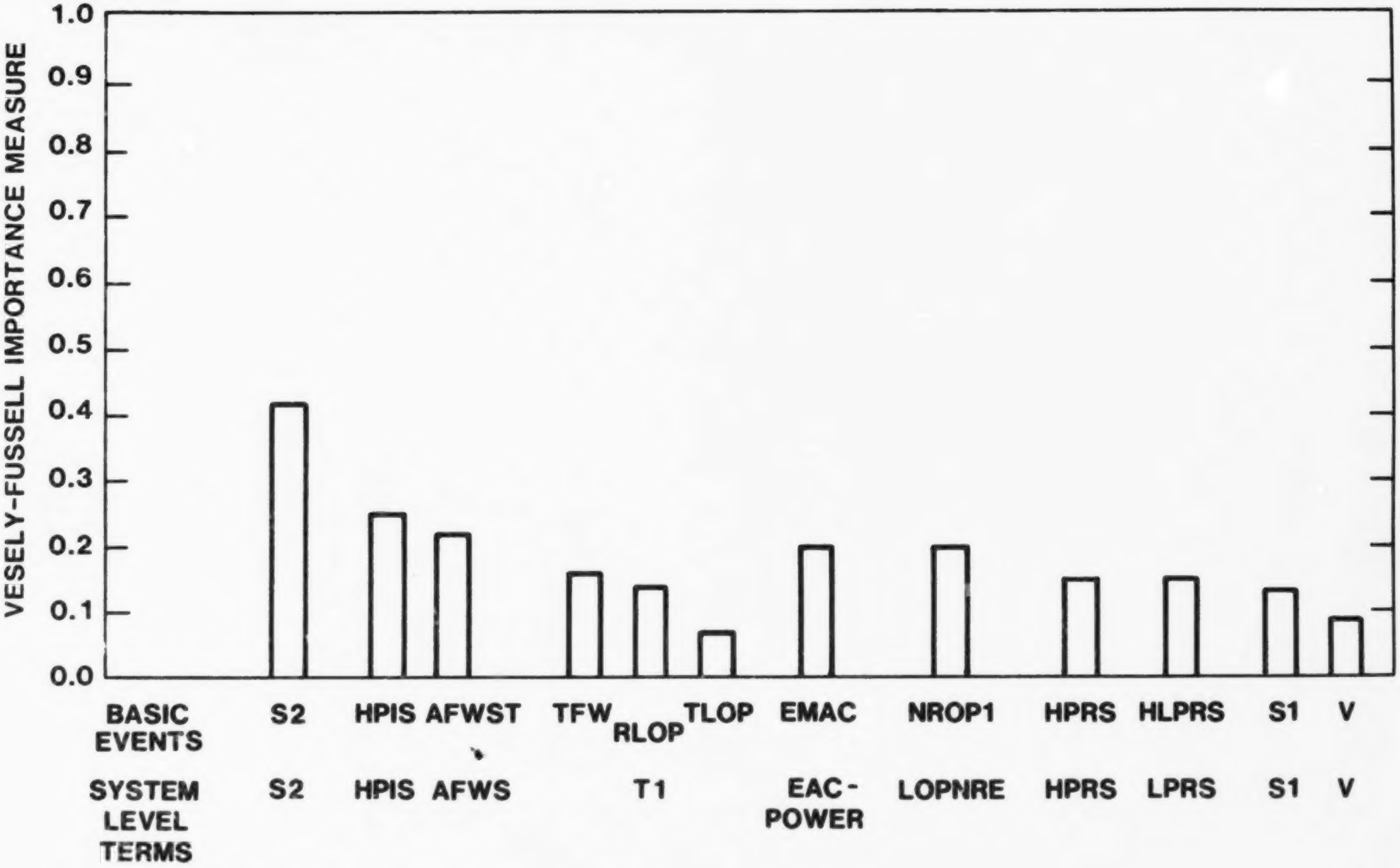

Figure 6. Surry, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

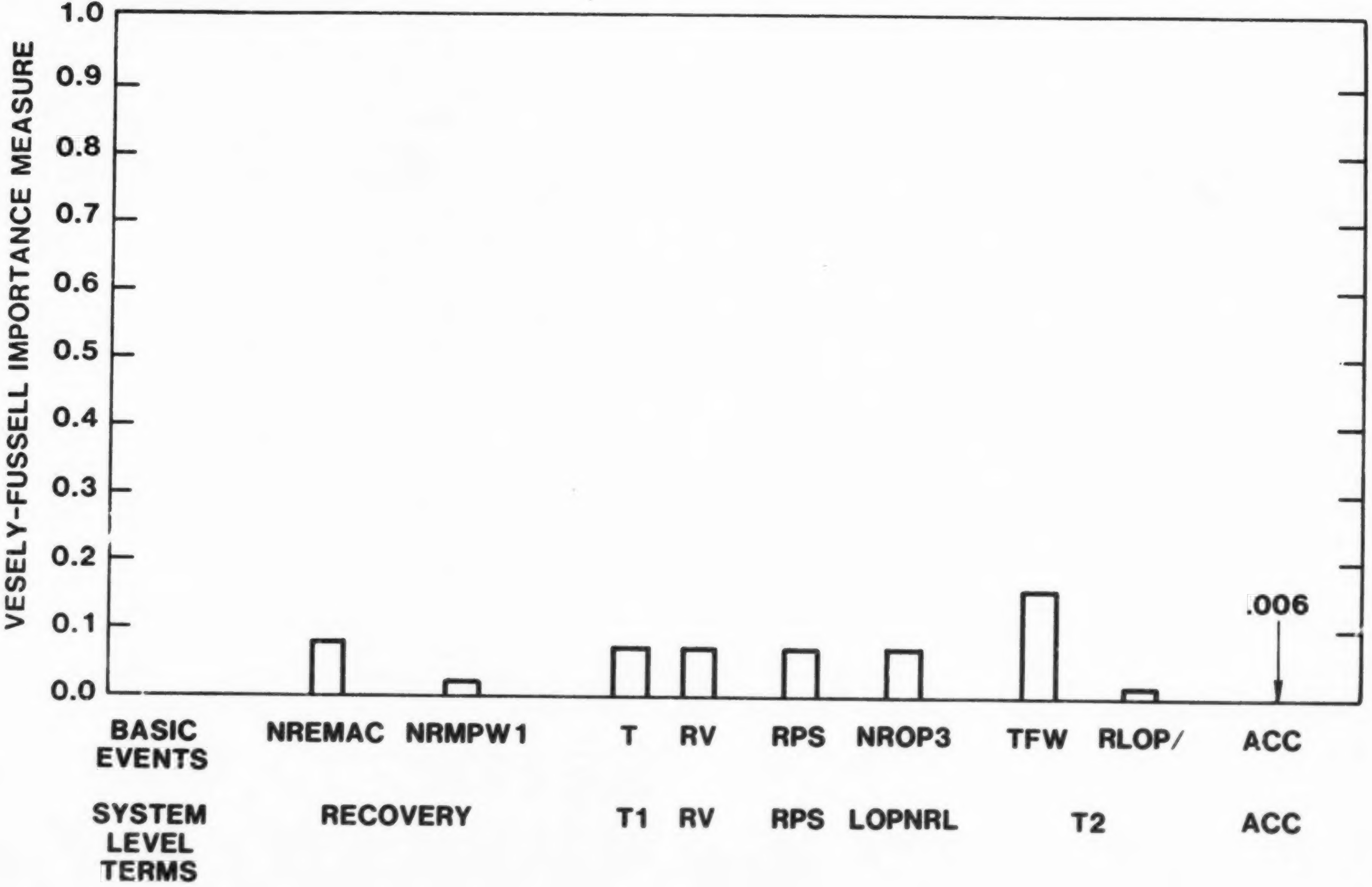

Figure 6. Surry, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

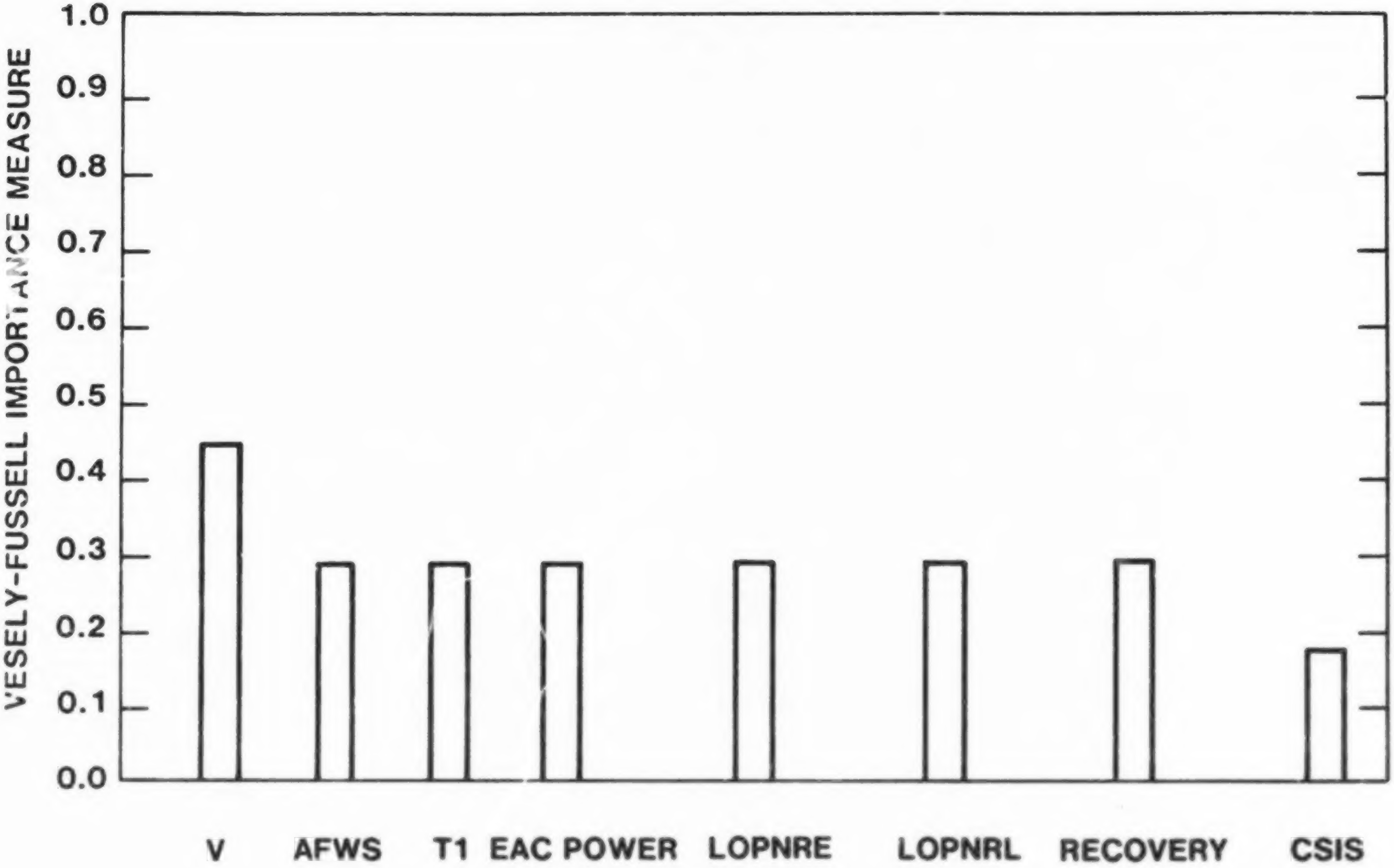

Figure 7. Surry, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

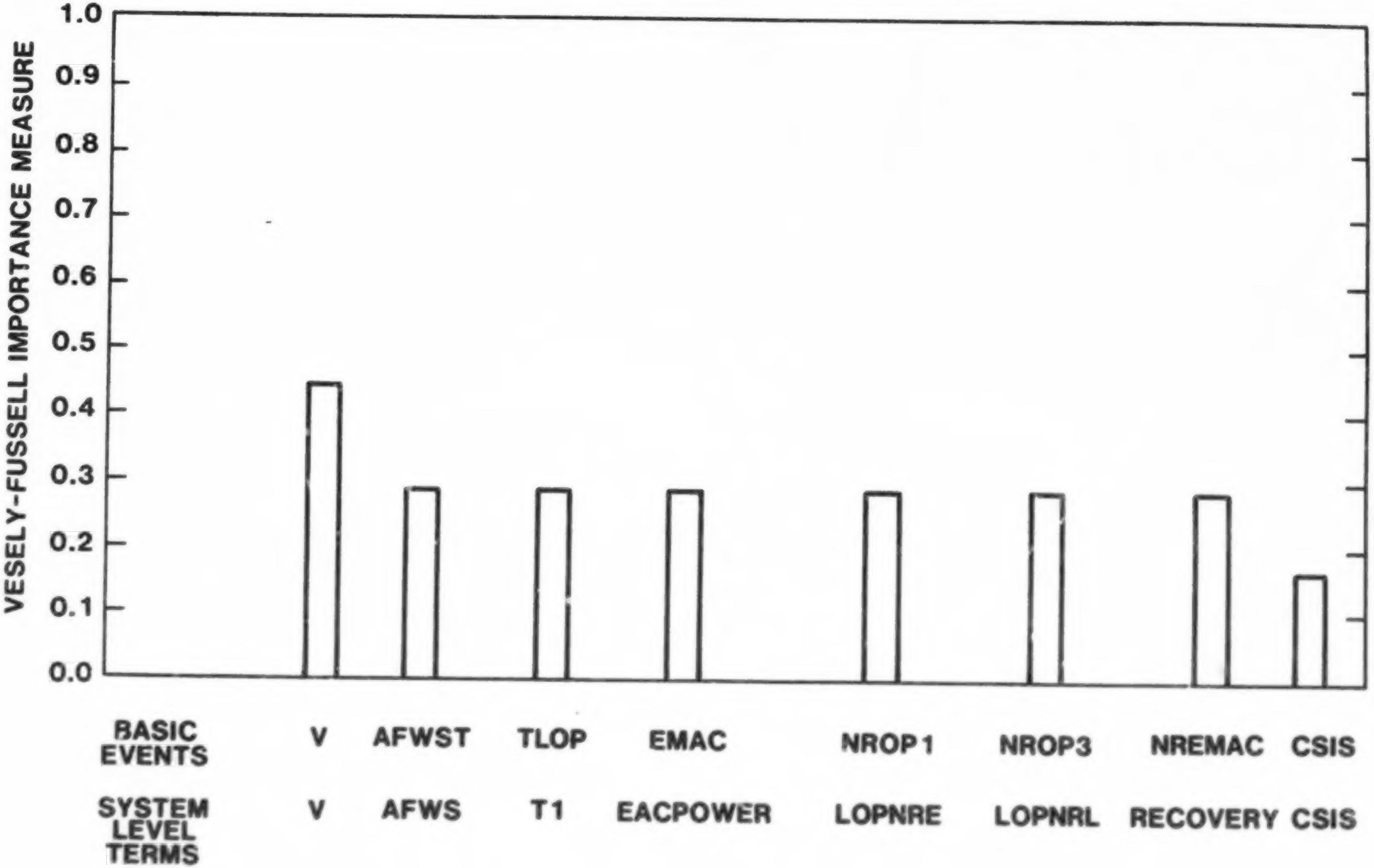

Figure 8. Surry, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

The Containment Spray Injection System (CSIS) is dominant with respect to risk, but not to core melt frequency. The CSIS is dominated by human error faults during test and maintenance procedures. Event V is the most dominant basic event. This follows from sequence V's probability of release of 1.0 in category 2.

c. Zion 1 and 2

The Zion analysis was done using the Zion Probabilistic Safety Study,[9] an independent study performed for Commonwealth Edison. Units 1 and 2 are considered to be exactly alike in the study, so the results of this analysis apply to both Zion units. Sequence fault trees or cut sets were not published in the Zion Study. Simplified fault trees were developed for this analysis. The trees were constructed using information on he sequences and plant systems from Table 8.10-3, and the appendices of Section 1.0 of the study. The letter report of the Review and Evaluation of the Zion Probabilistic Safety,[10] prepared by Sandia National Laboratories for the NRC, was also used extensively in studying the Zion sequences. It was of great value in facilitating understanding of the Zion Study. Most basic events were taken from Tables 1.5.2.3.4-13 and 1.5.2.3.4-16 of the study. The fault trees developed for Zion sequences are in Appendix A3 of this report. The sequence names on Table 22 are specific to this analysis, as the Zion Study did not lable sequences.

1. Criterion 1, Core Melt Frequency with Release

The Zion Study has different results for Criterion 1 than for Criterion 2. This is due to the study's contention that the containment will not fail in most sequences. In fact, only one sequence would fail the containment, as shown in Table 22. This sequence is the seismic event which results in loss of AC power, sequence 2 on Table 8.10-3 of the Zion study. A fault tree does exist for the sequence on page 7.2-27 of the Zion Study, which is reproduced here on Figure 9. However, no values were given for the basic events. The sequence was therefore modeled as a single event sequence in this analysis. Only 7 percent of the core melt frequency with release is not modeled by this event. The other sequences which fail the containment have frequencies much smaller than the seismic-loss of AC power event. The importance of this event is shown on Tables 25 and 26, and Figure 10.

2. Criterion 2, Total Core Melt Frequency

Four sequences dominate total core melt frequency. From Table 22, these are SLRCIRC, SESEIS, ALRCIRC, and MLRCIRC. These are sequences 1, 2, 3, and 4 of the Zion Study's

Table 22

Zion 1 and 2

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Westinghouse	4	Dry	1100	Independent

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures		
		1	2	3
1	SLRCIRC	-	.24	-
2	ALRCIRC	-	.07	-
3	MLRCIRC	-	.07	-
4	SESEIS	.93	.08	.93

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study PWR Release Categories							Sequence Frequency
		1	2	3	4	5	6	7	$(R\text{-}yr)^{-1}$
1	SLRCIRC								1.6E-5
2	SESEIS		1.0						5.6E-6
3	ALRCIRC								4.9E-6
4	MLRCIRC								4.9E-6
				Total Plant Core Melt Frequency					6.7E-5

Table 23

Zion 1 and 2 System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
OPERROR	Operator Error.	OPSWITCH SOPSWITCH SOPRWST CC9412AB
RMRS	Residual Heat Removal System.	CMRHRAB SMANHRA SMANHRB SMULTHW MANTRHR RHRAB
SESEIS	Seismic Event.	SESEIS
S_1	Medium LOCA (Small RSSMAP Loca).	M
A	Large LOCA.	A
S_2	Small LOCA (Small-Small RSSMAP LOCA).	S
SUMP	Containment Sump Blockages.	SUMP
HPRS	High Pressure Recirculation System.	MANSICH SMULTHW

Table 24

Zion
Boolean Terms from Sequence Fault Trees

Boolean Term	*Description	Value
A	Large LOCA	9.4E-4/R-yr
CC9412AB	Operator Failure to Open MOVs CC9412AB	8.5E-4
CMRHRAB	Common Mode Failures Between Trains A and B, Residual Heat Removal (RHR)System. Pumps or MOVs Fail to Start on Demand	8.3E-5
M	Medium LOCA	9.4E-4/R-yr
MANSICH	Charging or Safety Injection (SI) Pumps Under Maintenance and Hardware Failures	1.E-8
MANTRHR	Maintenance of One RHR Pump and Failure of the Other Train Given Large or Medium LOCA	9.8E-5
MULTHW	Multiple Hardware Failure with Large LOCA of Medium LOCA	1.09E-4
OPSWICH	Failure of Operator to Switch Over to Recirculation Phase or Stop Refueling Water Storage Tank (RWST) Pump at Low-Low Level Given Medium or Large LOCA Sequence	4.0E-3
RHRAB	Common Mode Failure of RHR Trains A and B Given Large or Medium LOCA	7.6E-5
S	Small LOCA	3.5E-2/R-yr
SESEIS	Seismic Event Resulting in Core Melt.	5.6E-6/R-yr
SMANRHRA	RHR Pump A Under Maintenance and Hardware Failures with Small LOCA	6.3E-5
SMANRHRB	RHR Pump B Under Maintenance and Hardware Failures with Small LOCA	6.3E-5

Table 24 (Cont.)

Zion
Boolean Terms from Sequence Fault Trees

Boolean Term	Description	Value
SOPSWICH	Failure of Operator to Initiate Switchover to Recirculation with Small LOCA	1.3E-4
SOPRWST	Failure of Operator to Stop Pumps at RWST Low-Low Level Alarm with Small LOCA	3.3E-5
SUMP	Containment Sump Blockage	2.9E-5

*Descriptions of most Boolean terms are taken from Tables 1.5.2.3.4.13 and 1.5.2.3.4-16 of the Zion Probabilistic Safety Study.

Table 25

Zion

Contribution of Front Line and Support Systems to Total Core Melt Frequency With Release: Criterion 1 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
SEIS	.93	1.0	1.0

Table 26

Zion

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency With Release: Criterion 1 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
SEIS	SEIS	.93	1.0	1.0

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Table 27

Zion

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
OPERROR	.23	.77	.50
S_2	.22	.77	.50
RHRS	.10	.65	.22
SEIS	.08	.63	.18
S_1	.07	.62	.16
A	.07	.62	.16
HPRS	.03	.58	.07
SUMP	.01	.56	.03

Table 28

Zion

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criterion 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
OPERROR	OPSWITCH	.11	.66	.25
	SOPSWITCH	.07	.61	.15
	SOPRWST	.02	.56	.04
	CC9412AB	.02	.56	.04
	MULTHW	.003	.55	.007
	SMULTHW	5.2E-6	.54	1.2E-5
S_2	S	.22	.77	.50
RHRS	SMANRHRA	.04	.59	.10
	SMANRHRB	.03	.58	.07
	CMRHRAB	.02	.56	.03
	MULTHW	.003	.55	.007
	MANTRHR	.003	.55	.006
	RHRAB	.002	.55	.005
	SMULTAW	5.2E-6	.55	1.2E-5
SEIS	SEIS	.08	.63	.18
S_1	M	.07	.62	.16
A	A	.07	.62	.16
HPRS	MANSICH	.03	.58	.07
	SMULTAW	5.2E-6	.55	1.2E-5
SUMP	SUMP	.01	.56	.03

Table 29

Zion

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
SEIS	.93	1.0	1.0

Table 30

Zion

Contribution of Basic Events, Grouped by Front Line Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
SEIS	SEIS	.93	1.0	1.0

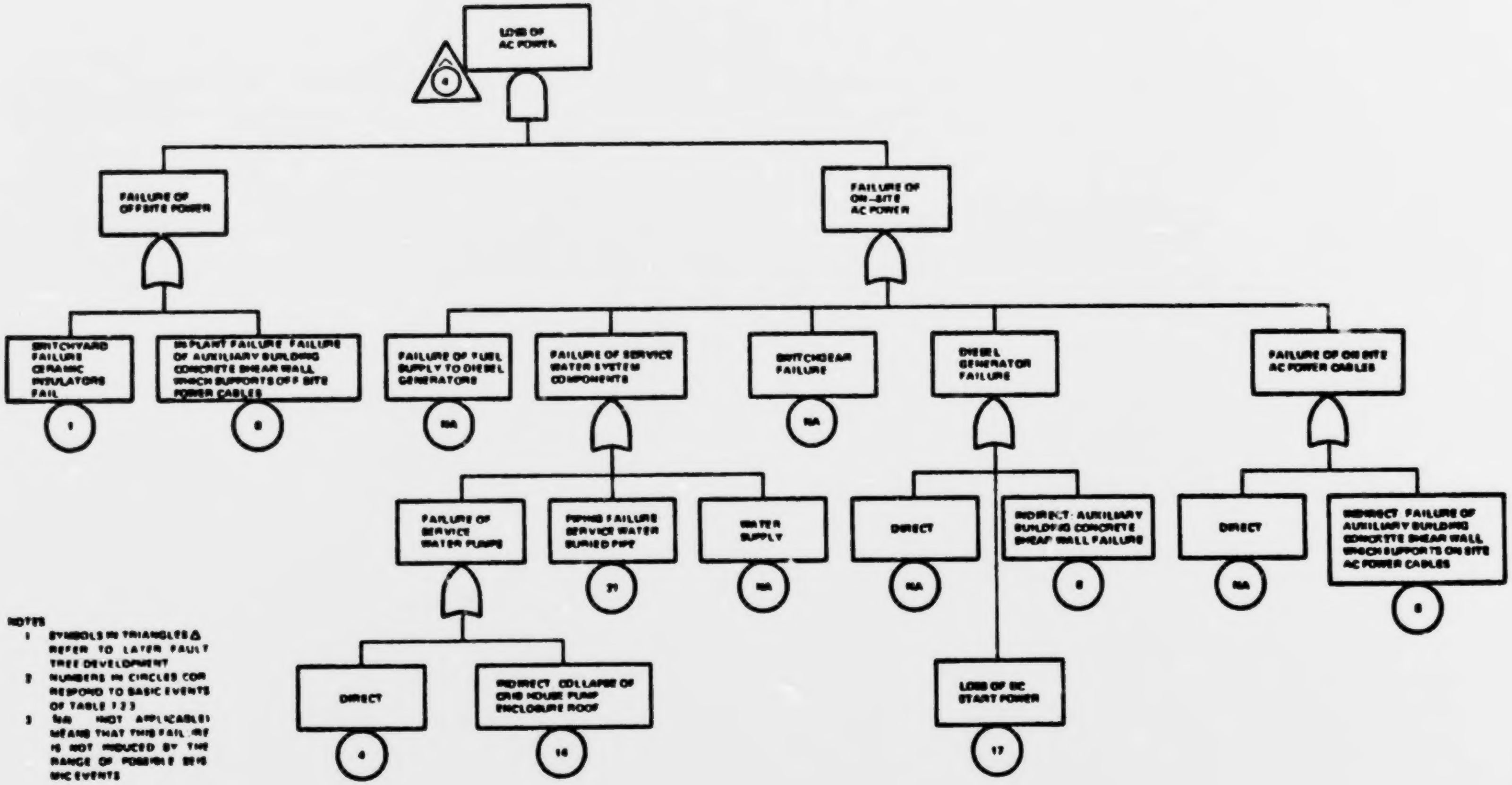
LOSS OF AC POWER
4
FAILURE OF OFFSITE POWER
FAILURE OF ON-SITE AC POWER
SWITCHYARD FAILURE CERAMIC INSULATORS FAIL
1
INPLANT FAILURE FAILURE OF AUXILIARY BUILDING CONCRETE SHEAR WALL WHICH SUPPORTS OFF SITE POWER CABLES
8
FAILURE OF FUEL SUPPLY TO DIESEL GENERATORS
NA
FAILURE OF SERVICE WATER SYSTEM COMPONENTS
SWITCHGEAR FAILURE
NA
DIESEL GENERATOR FAILURE
FAILURE OF ON SITE AC POWER CABLES
FAILURE OF SERVICE WATER PUMPS
PIPING FAILURE SERVICE WATER BURIED PIPE
21
WATER SUPPLY
NA
DIRECT
NA
INDIRECT: AUXILIARY BUILDING CONCRETE SHEAR WALL FAILURE
8
DIRECT
NA
INDIRECT FAILURE OF AUXILIARY BUILDING CONCRETE SHEAR WALL WHICH SUPPORTS ON SITE AC POWER CABLES
8
DIRECT
4
INDIRECT COLLAPSE OF CRIB HOUSE PUMP ENCLOSURE ROOF
16
LOSS OF DC START POWER
17
NOTES
1 SYMBOLS IN TRIANGLES △ REFER TO LATER FAULT TREE DEVELOPMENT
2 NUMBERS IN CIRCLES CORRESPOND TO BASIC EVENTS OF TABLE 7.2.3
3 NA (NOT APPLICABLE) MEANS THAT THIS FAILURE IS NOT INDUCED BY THE RANGE OF POSSIBLE SEISMIC EVENTS

Figure 9. Zion Seismic Core Melt Fault Tree--Branch Expansion, Loss of AC Power[9]

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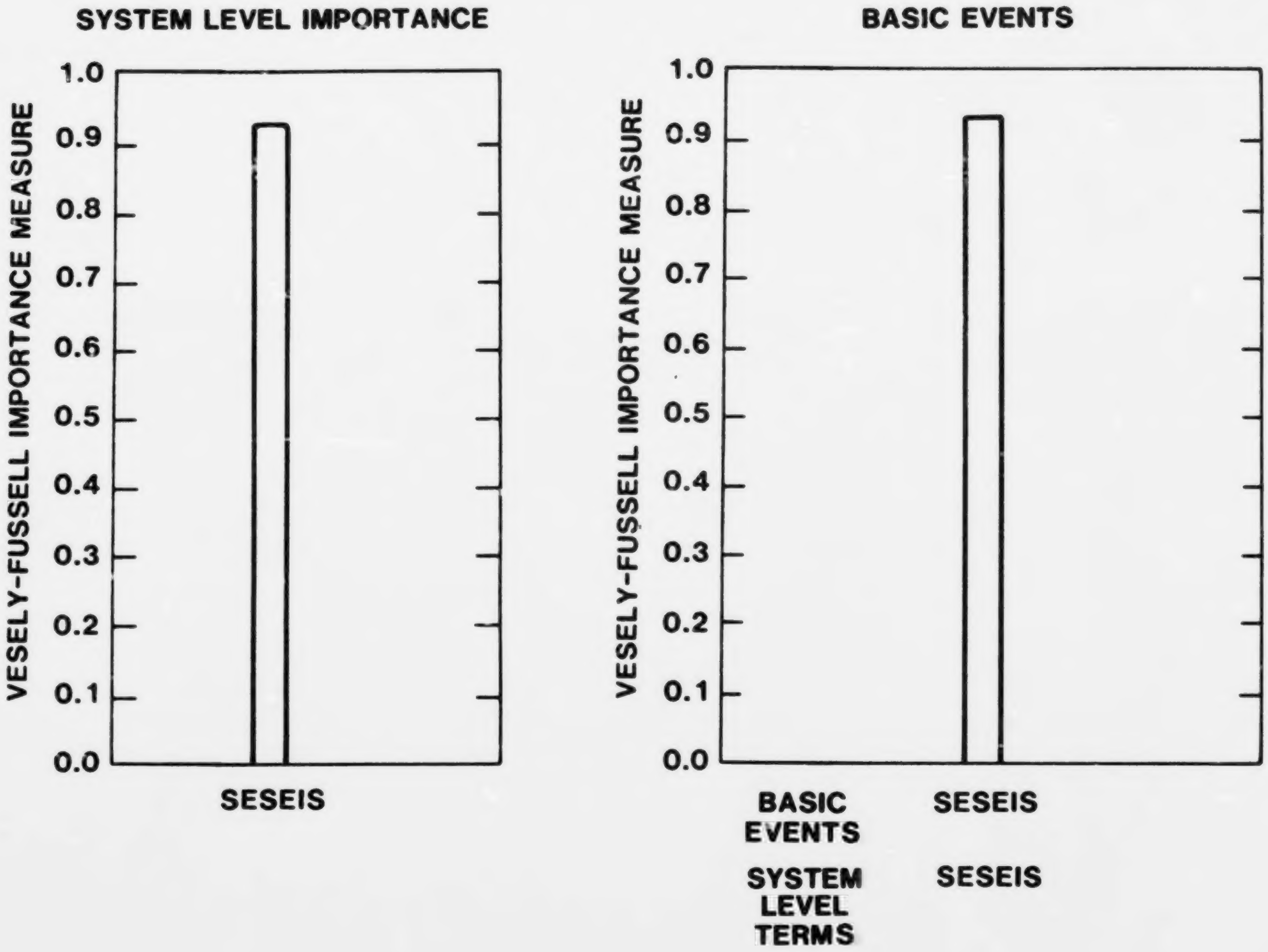

Figure 10. Zion, Importance of Front Line and Support Systems and Basic Events to Total Core Melt Frequency with Release: Criterion 1

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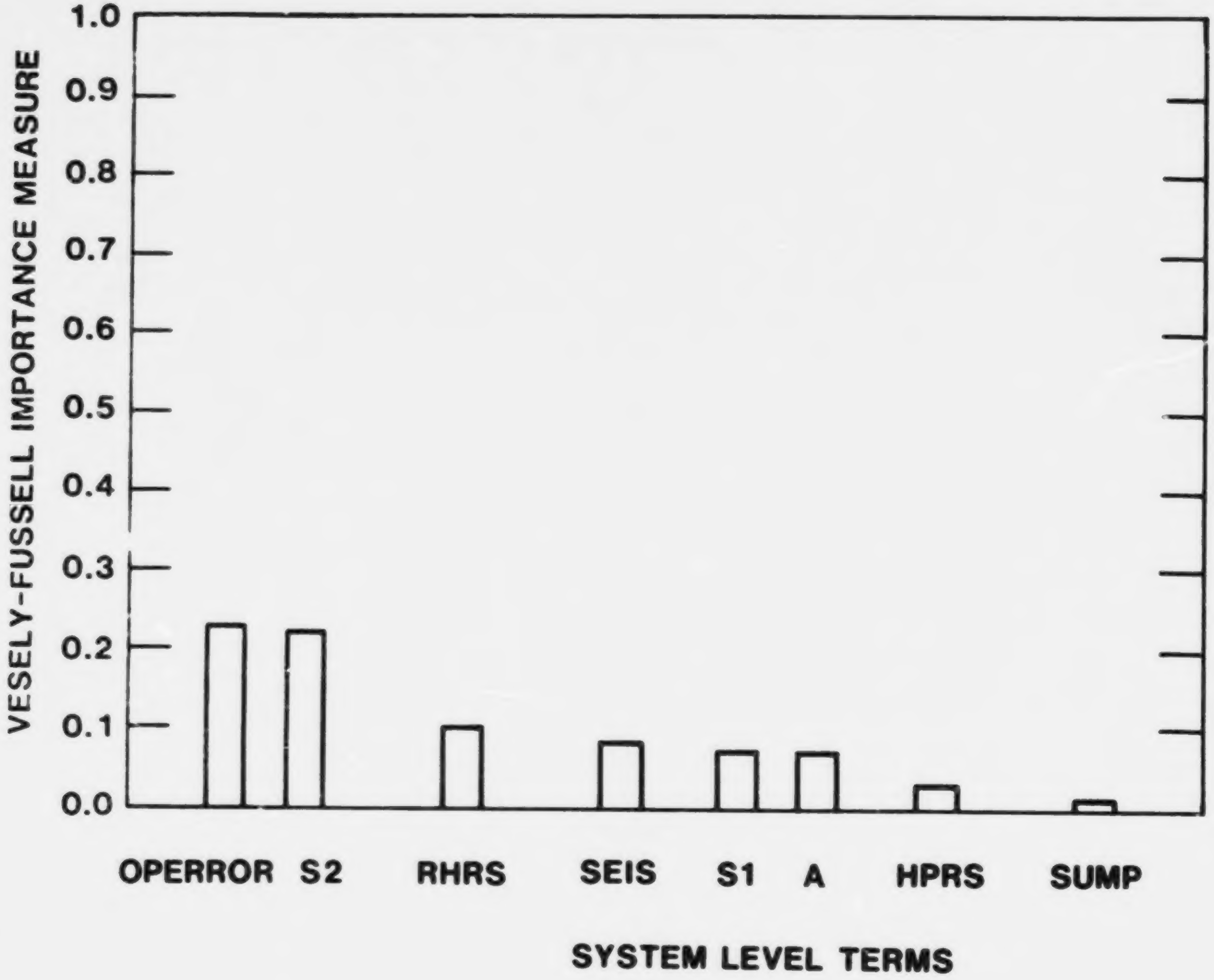

Figure 11. Zion, Importance of Front Line and Support System to Total Core Melt Frequency: Criterion 2

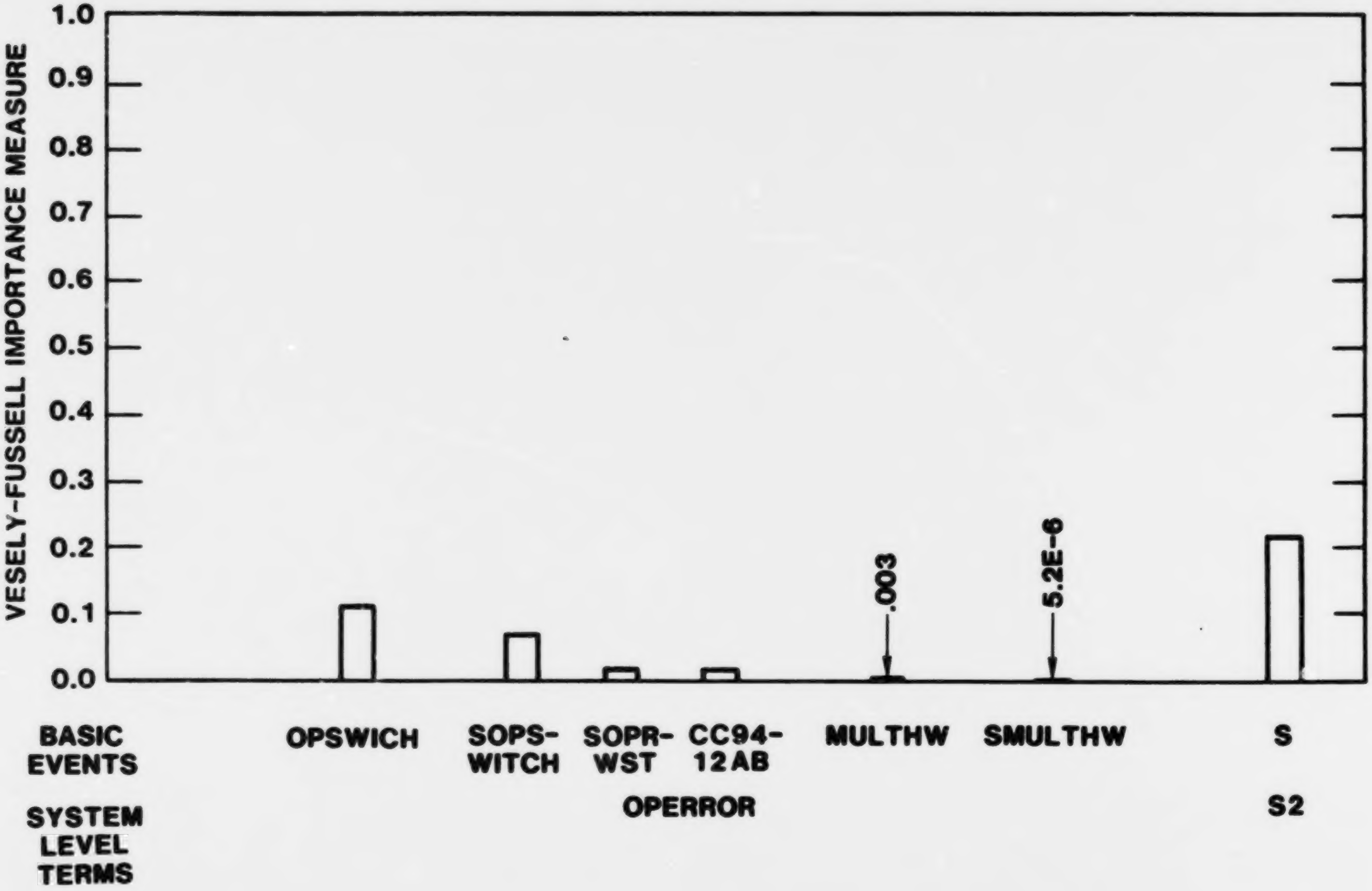

Figure 12. Zion, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criterion 2

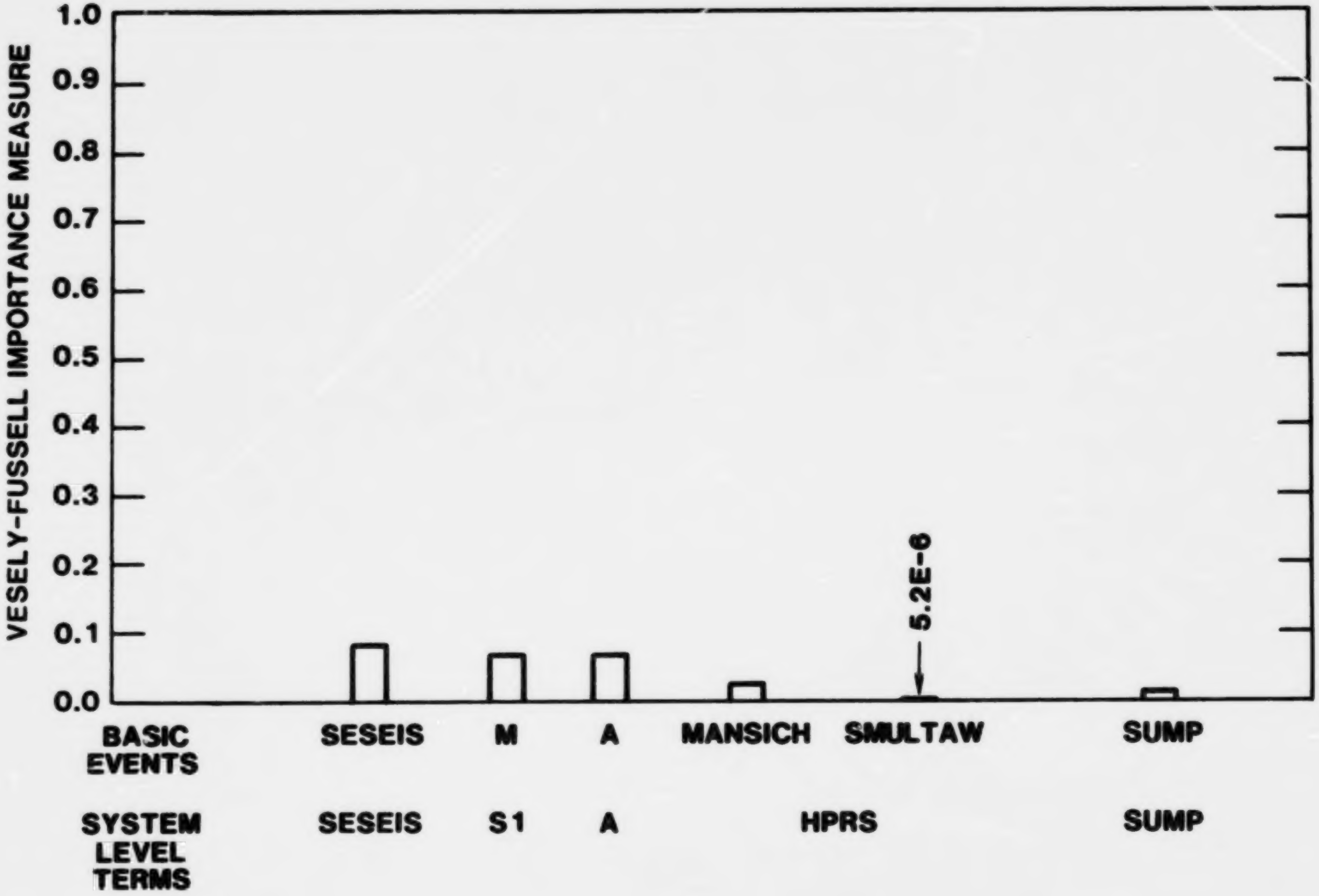

Figure 12. Zion, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criterion 2 (Cont)

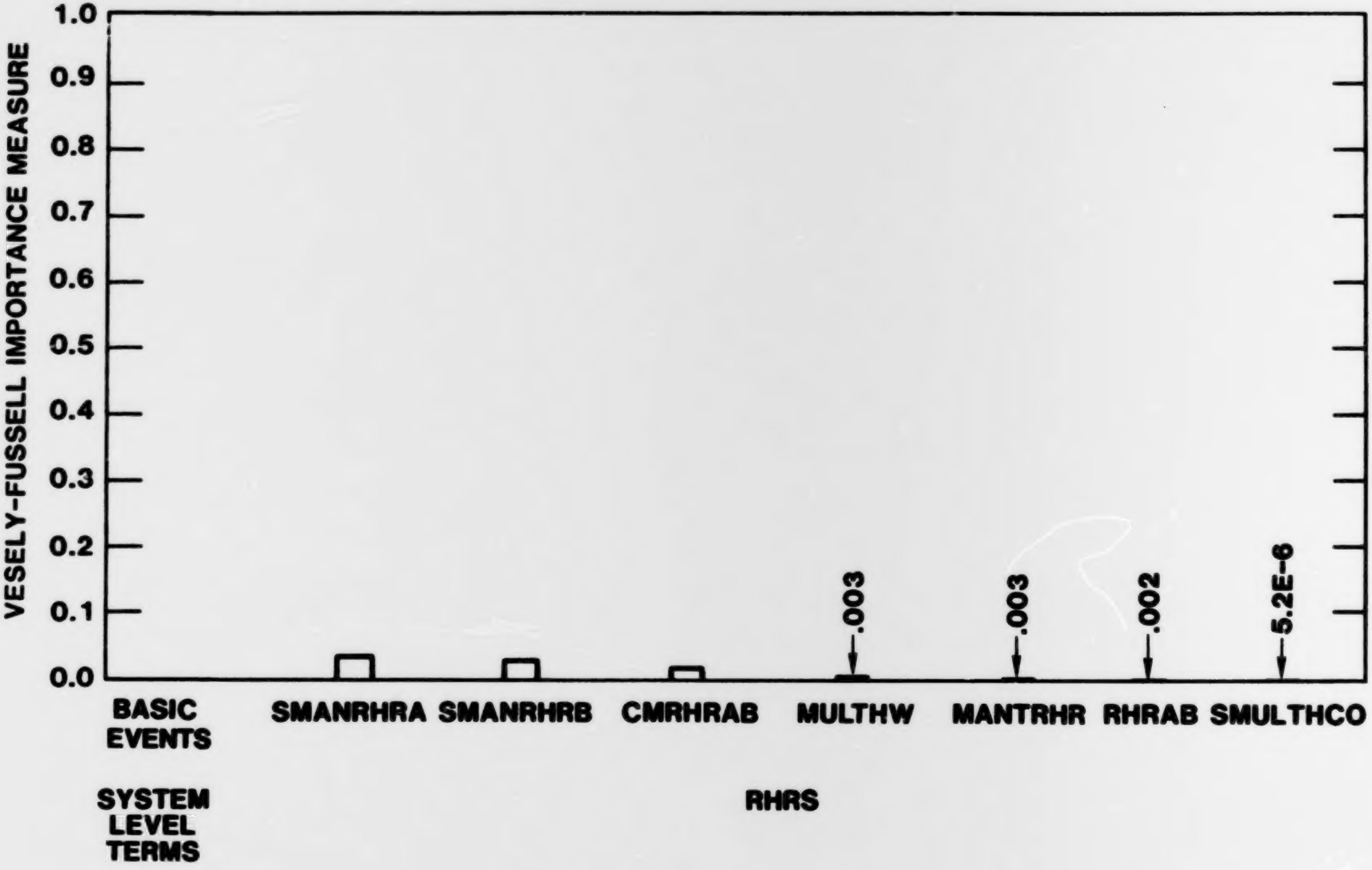

Figure 12. Zion, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criterion 2 (Cont)

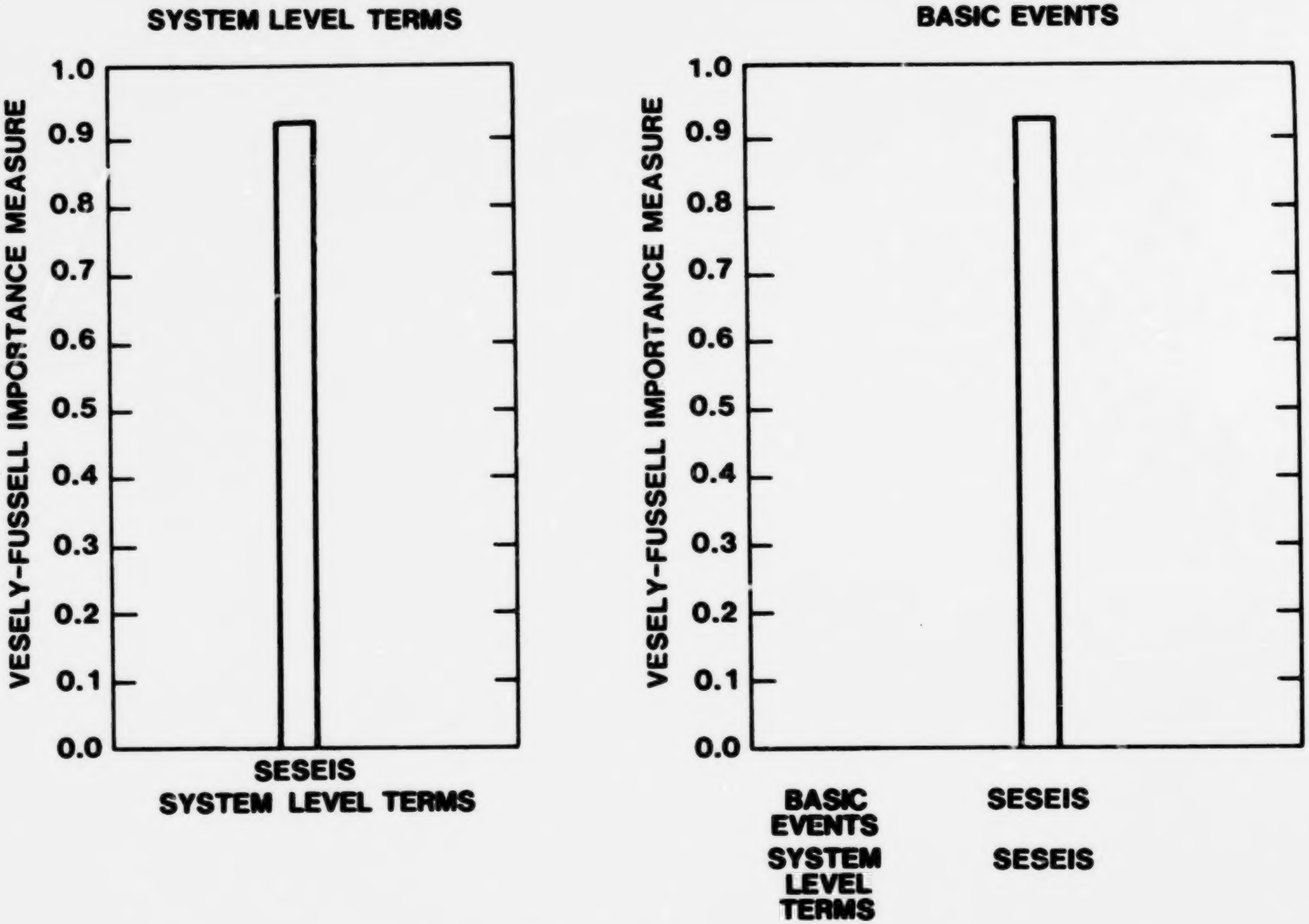

Figure 13. Zion, Importance of Front Line and Support Systems, and Basic Events to Total Plant Risk: Criterion 3

Table 8.10-7. The cut sets for these sequences only model 45 percent of the core melt frequency. The cut off point of 5 percent of total core melt frequency for dominant sequences for this analysis may significantly impact the Zion results. However, the sum of all the sequence frequencies on Table 8.10-7 account for only 66 percent of the total core melt frequency reported in the Zion Study. So, this analysis is reasonably close to the limits allowed by the information provided in the Zion Study.

The importances of the system level terms are listed in Table 27 and shown on Figure 11. The systems' basic event importances are on Table 28 and Figure 12. Operator error dominates the system level terms, followed by the Residual Heat Removal Systems (RHRS) and the single event sequence SESEIS. The two most significant human errors (OPSWICH and SOPSWICH) involve operator failure to switch over to recirculation phase at the proper time. The basic event value of OPSWICH is larger than SOPSWICH, reflecting the stressful environment of medium and large LOCAs. RHRS failure is dominated by SMANRHRA and SMANRHRB. These events are rather vaguely defined in the Zion Study, but basically involve failure of the RHR pumps A and B. All the cut sets for the Zion sequences (A3) are two event cut sets. Thus, all basic events are independent of each other.

3. Criterion 3, Total Plant Risk

The results for Criterion 3 are exactly the same as the results for Criterion 1. The importance of SESEIS is listed on Tables 29 and 30 and shown in Figure 13.

d. Indian Point 2

The analysis for Indian Point 2 was done using the Indian Point Probabilistic Safety Study.[9] This study was done on Units 2 and 3. It is an independent duty done for Consolidated Edison. Sequence fault trees and cut set information were not published in the Indian Point Study. Simplified fault trees were constructed for the dominant sequences, and are given in Appendix A5. The fault trees and basic events were developed using information from Table 8.3-19, the appendices of Section 1.0 of the study, and the Review and Evaluation of the Indian Point Probabilistic Safety Study, NUREG/CR-2934, prepared for the NRC by Sandia National Laboratories, which was very useful in facilitating understanding of the Indian Point Study. Most basic events were developed from the discussion of the dominant sequences in the letter report. The sequences are listed on Table 31.

1. Criterion 1: Total Core Melt Frequency with Release

The three dominant sequences for Criterion 1 involve sequences for external events. The sequences are a seismic event resulting in loss of AC power (SESEIS), fire in the electrical tunnel or switchgear room (SEFIRE), and loss of all AC power due to hurricane winds (SEHUR). Fault trees for the seismic and hurricane sequences were published in the Indian Point Study. However, the values of the basic events of these trees were not published in the study. So these sequences were modeled as single event sequences, based on the discussion of these sequences in Sandia's review of the study. These three sequences account for 94 percent of the frequency of core melt with release of radioactivity into the environment.

The system level terms on Table 32 are really not systems, but the type of event which initiate the sequences. Only the fire sequence is modeled with more than one event. The importances of the basic events are listed on Table 35 and shown on Figure 14. The primary hazards in the seismic and hurricane events are loss of offsite power due to the intensity of the event and collapse of Unit 1's superheater smoke stack onto Unit 2's control room. The fire sequence involves a fire in one of three locations, the PAB end of the electrical tunnel, the CB end of the electrical tunnel, on the switchgear room. A fire in any of these locations could fail power to the charging pumps, containment spray pumps, auxiliary feedwater system, safety injection pumps, and component cooling water pumps. A small LOCA through the reactor coolant pump seals would result. The basic events of these sequences are independent of each other.

2. Criterion 2, Total Core Melt Frequency

Table 31 shows five sequences are dominant to core melt frequency. The cut sets of these sequences account for 83 percent of the core melt. In addition to the three external event sequences dominant to Criterion 1, another fire sequence (SLFFIR) and a turbine trip sequence (SEFCTT) are dominant. The fire sequence involves a fire in the electrical tunnel right stack, which would not fail auxiliary feedwater and high pressure injection systems, resulting in no release of radioactive material into the environment. Sequence SEFCTT is a turbine trip due to loss of offsite power followed by failure of the emergency AC system.

The importance of the system level terms (Table 36 and Figure 15) show that emergency AC power is the only plant system important to core melt frequency,. The basic event importances (Table 37 and Figure 16) show that the fire

Table 31

Indian Point 2

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Westinghouse	4	Dry	873	Independent

Dominant Accident Sequences

Sequence	Contribution to Overall Criteria Impact Measures		
	1	2	3
1 SESEIS	.42	.30	.42
2 SEFIRE	.42	.30	.42
3 SEHUR	.08	.06	.08
4 SEFCTT	-	.06	-
5 SEHUR	-	.06	-

Dominant Accident Sequence Release Category Probabilities

Sequence	Reactor Safety Study PWR Release Categories							Sequence Frequency
	1	2	3	4	5	6	7	$(\text{R-yr})^{-1}$
1 SESEIS		1.0						1.4E-4
2 SEFIRE		1.0						1.4E-4
3 SLFFIR								5.0E-5
4 SEFCTT								3.0E-5
5 SEHUR		1.0						2.7E-5

Total Plant Core Melt Frequency 4.7E-4

Table 32

Indian Point 2
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
EACPOWER	Emergency AC Power System	FREP5A FREP6A
FIRE	Fire Induced Events	EICB ETPAB ETRSCB ETRSPAB SWGEAR
HURRICANE	Hurricane Induced Events	HUR
SESEIS	Loss of Offsite Power Due to Seismic Events	LOP
TI	Turbine Trip Due to Loss of Offsite Power	TT

Table 33

Indian Point 2
Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
ETCB	Fire in Electrical Tunnel, CB End	3.2E-5
ETPAB	Fire in Electrical Tunnel, PAB End	5.6E-5
ETRSCB	Fire in Electrical Tunnel Right Stack, CB End	2.4E-5
ETRSPAB	Fire in Electrical Tunnel Right Stack, PAB End	2.4E-5
FREP5A	Power is Lost to Buses 2A, 3A, 5A, for 60 Minutes Within 6 Hours Following TT Initiating Event	8.9E-5
FREP6A	Power is Lost to Buses 2A, 3A, 6A, for 60 Minutes Within 6 Hours Following TT Initiating Event.	8.9E-5
HUR	Hurricane Initiating Core Melt Sequence	2.7E-5/R-yr
LOP	Loss of Power	1.4E-4/R-yr
SWGEAR	Fire in Switchgear Room	5.6E-5
TT	Turbine Trip Due to Loss of Offsite Power	.18/R-yr

Table 34

Indian Point 2

Contribution of Front Line and Support Systems to Total Core Melt Frequency With Release: Criterion 1 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
FIRE	.43	.50	.46
SEIS	.42	.48	.45
HURRICANE	.08	.14	.09

Table 35

Indian Point 2

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency With Release: Criterion 1 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
FIRE	ETPAB	.17	.23	.18
	SWGEAR	.17	.23	.18
	ETCB	.10	.16	.10
SEIS	LOP	.42	.48	.45
HURRICANE	HUR	.08	.14	.09

Table 36

Indian Point 2

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
FIRE	.41	.58	.49
SEIS	.30	.47	.36
T_1	.07	.23	.08
EACPOWER	.07	.23	.08
HURRICANE	.06	.23	.07

Table 37

Indian Point 2

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency With Release: Criterion 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
FIRE	ETPAB	.12	.30	.14
	SWGEAR	.12	.30	.14
	ETCB	.07	.23	.08
	ETRSPAB	.05	.22	.06
	ETRSCB	.05	.22	.06
SEIS	LOP	.30	.47	.36
T_1	TT	.07	.23	.08
EACPOWER	FREPGA	.03	.20	.04
	FREP5A	.03	.20	.04
HURRICANE	HUR	.06	.23	.07

Table 38

Indian Point 2

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
FIRE	.43	.50	.46
SEIS	.42	.49	.45
HURRICANE	.08	.15	.09

Table 39

Indian Point 2

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
FIRE	ETPAB	.17	.24	.18
	SWGEAR	.17	.24	.18
	ETCB	.10	.17	.10
SEIS	LOP	.42	.49	.45
HURRICANE	HUR	.08	.15	.09

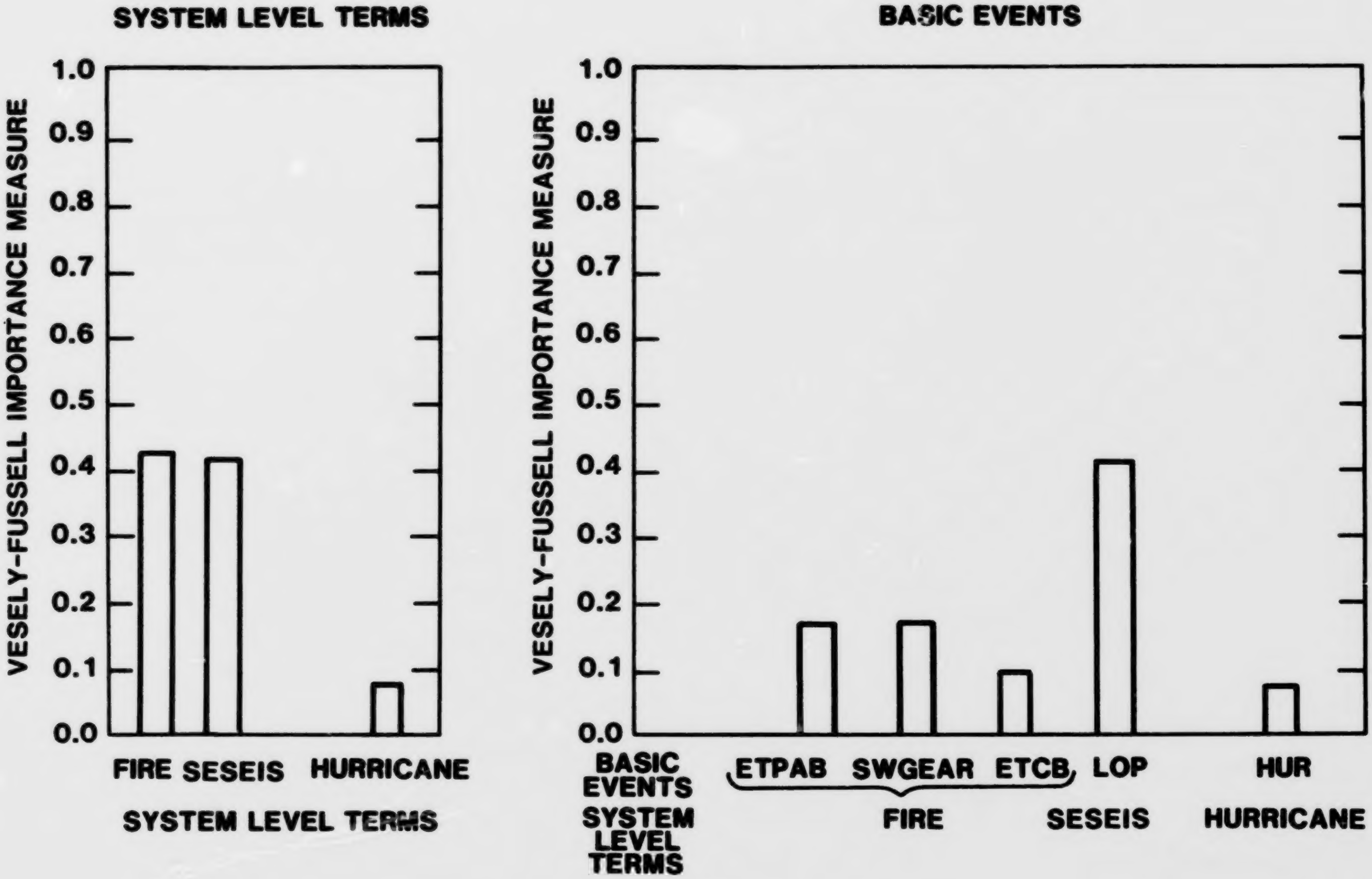

Figure 14. Indian Point 2, Importance of Front Line and Support Systems, and Basic Events, to Total Core Melt Frequency: Criterion 1

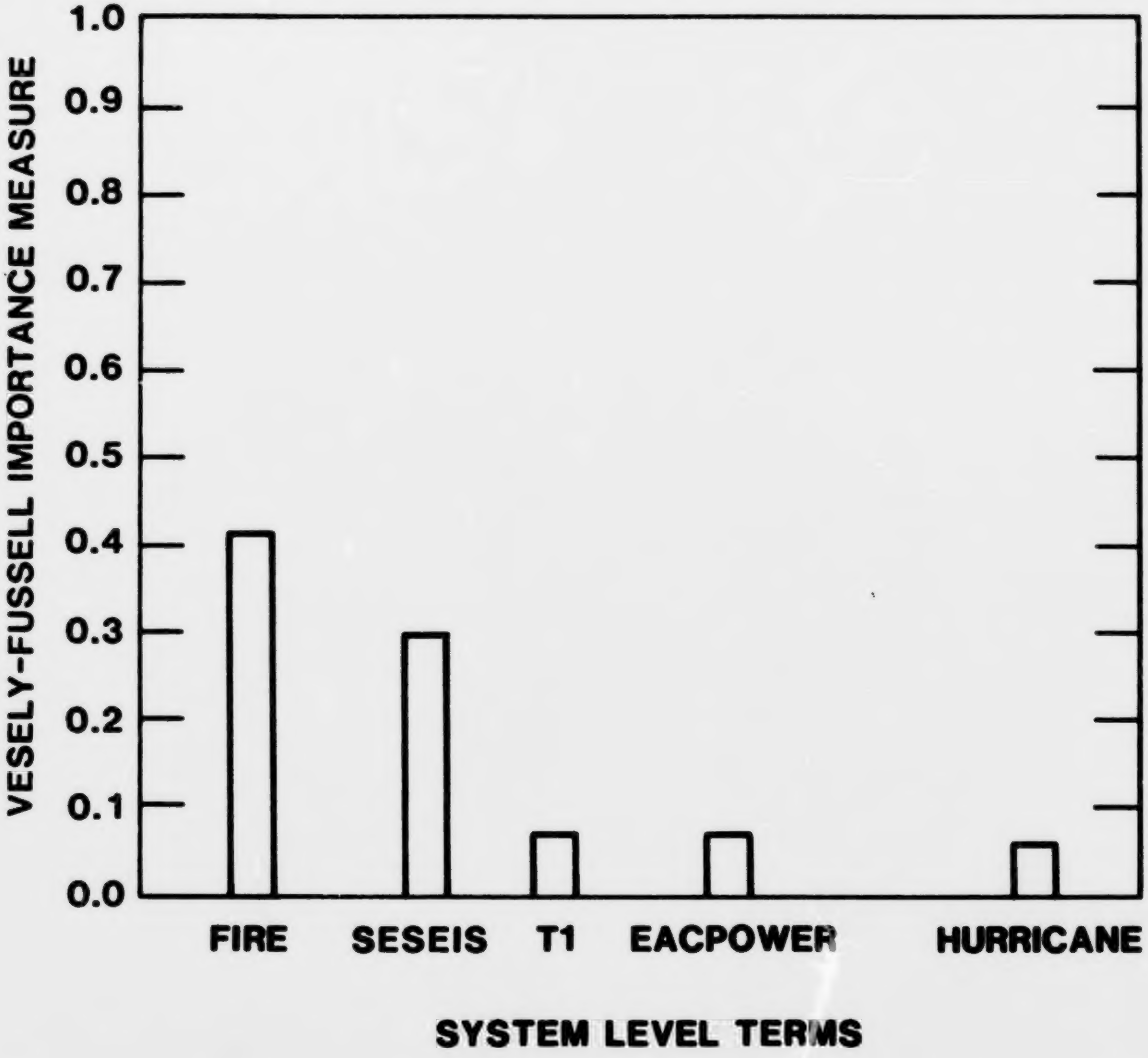

Figure 15. Indian Point 2, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2

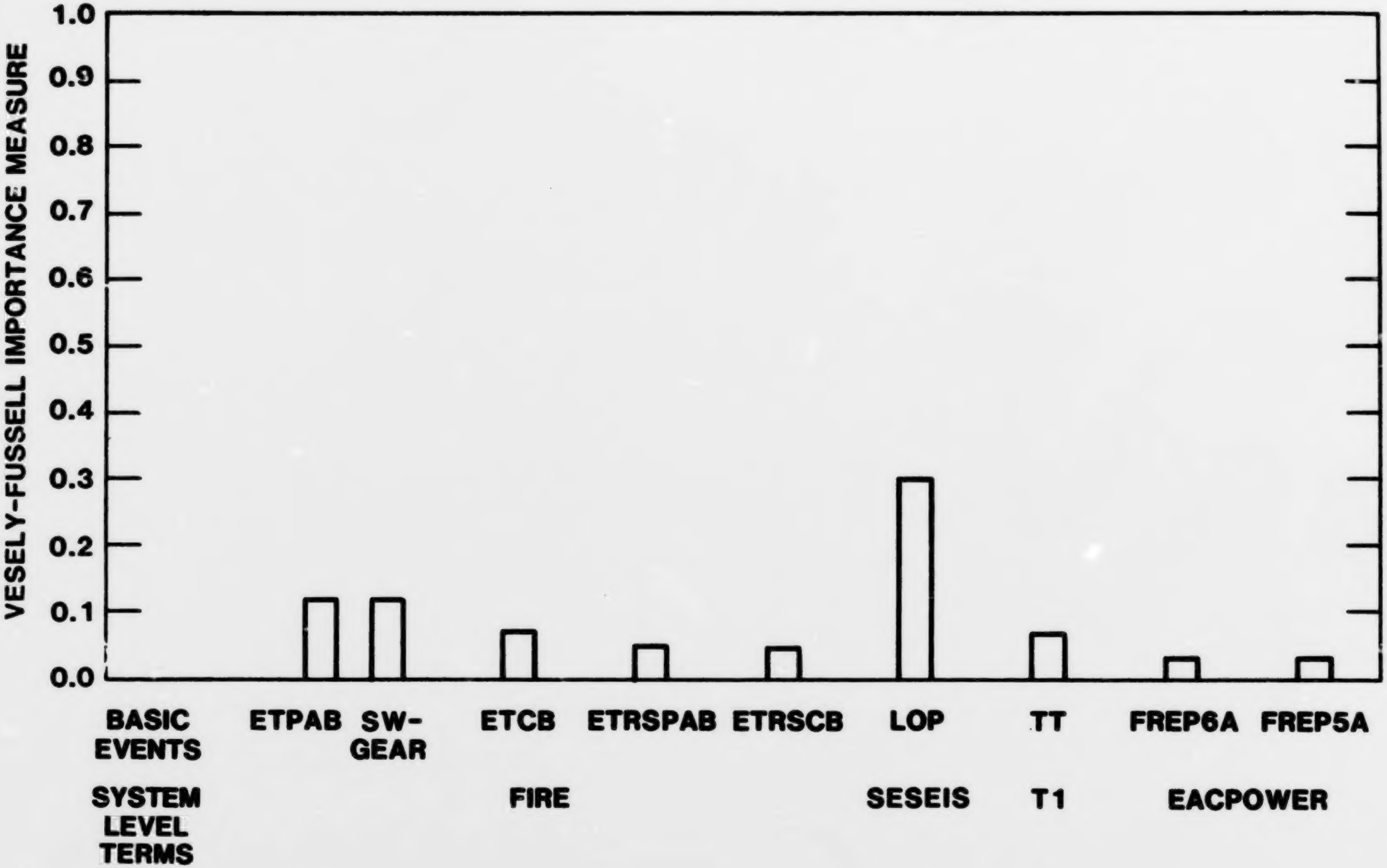

Figure 16. Indian Point 2, Importance of Basic Events Grouped by Systems, to Total Core Melt Frequency: Criterion 2

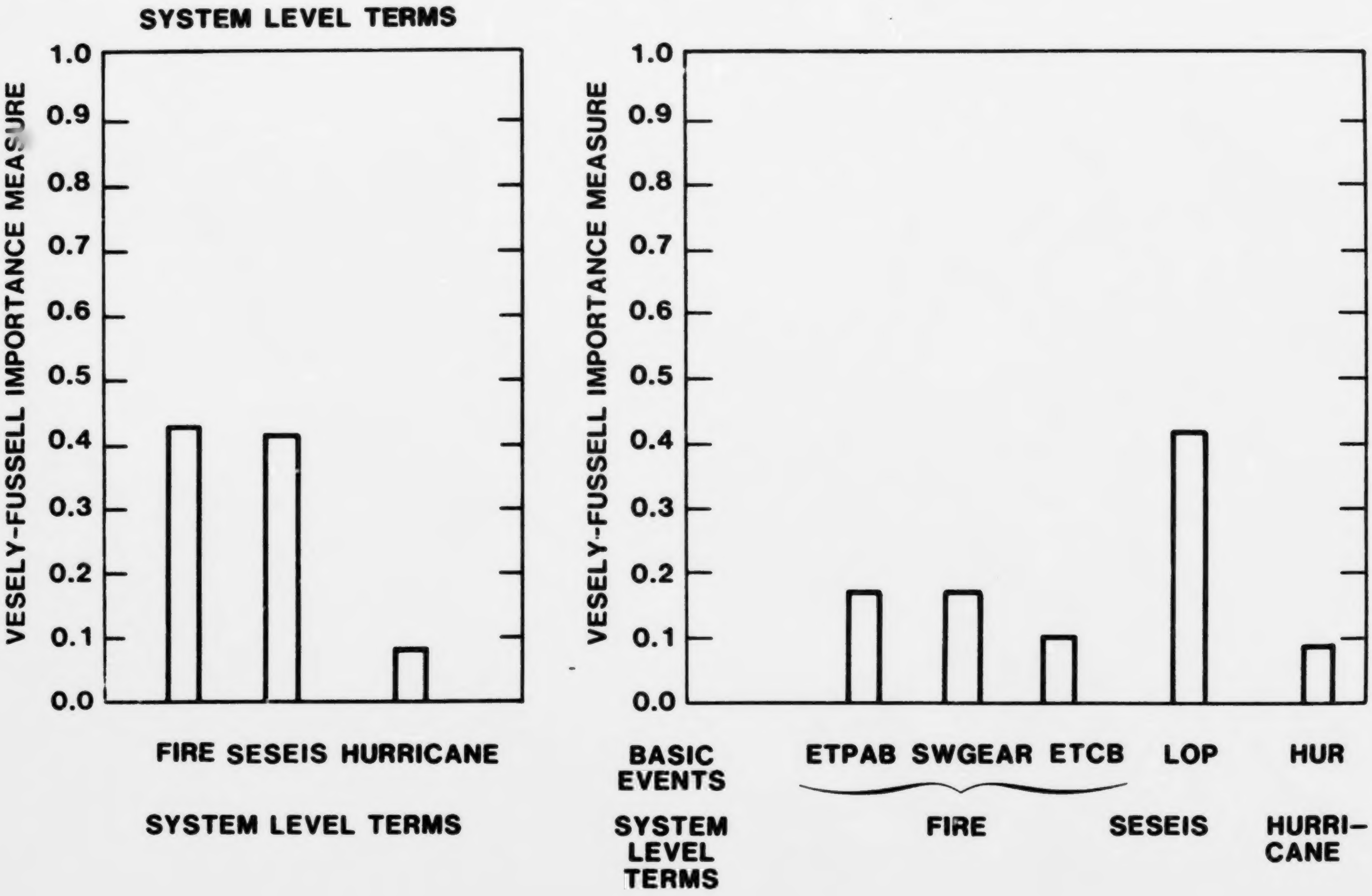

Figure 17. Indian Point 2, Importance of Front Line and Support Systems, and Basic Events, to Total Plant Risk: Criterion 3

events are dominated by fire in the PAB end of the electrical tunnel and the switchgear room. All the basic events are independent of each other.

3. Criterion 3, Total Plant Risk

The results for Criterion 3 are virtually identical to the results for Criterion 1, as the three sequences which fail the containment all have a release probability of 1.0 in a high risk release category. The results are on Tables 38 and 39, and Figure 17.

e. Indian Point 3

The analysis on Indian Point 3 was done using the Indian Point Probabilistic Study[9] in a manner similar to the analysis for Indian Point 2. The simplified fault trees for the sequences are given in Appendix A5. Most of the basic events were taken directly from Tables 1.6.2.3.4-13 and 1.6.2.3.4-19 of the Indian Point Study, and are listed on Tables 41 and 42.

1. Criterion 1, Frequency of Core Melt with Release

Table 40 shows only one sequence (SEFIRE) is dominant to Criterion 1. This sequence is similar to the fire sequences for Indian Point 2. A fire in either the switchgear room or the cable spreading room could fail power to the charging pumps, the containment spray pumps, the component cooling pumps, and the safety injection pumps. A small LOCA in the reactor coolant pump seals would result. Loss of the containment sprays and fans would result in containment failure. The importances are listed on Tables 43 and 44, and shown on Figure 18.

2. Criterion 2, Total Core Melt Frequency

Four sequences are dominant to core melt frequency. The cut sets developed for these sequences model 85 percent of the core melt frequency. The four sequences, SLFHPRC, SEFIRE, LLLPRC, and MLLPRC are sequences 1, 2, 3, and 4, respectively, on Table 8.3-10 of the Indian Point Study. The sequences other than the fire sequence involve failure of the recirculation systems, either in the low pressure mode or high pressure mode.

The system level results (Table 45, Figure 19) show the recirculation system as the only plant system dominant to core melt frequency. Operator error is also significant. The recirculation system (RCIR) is described as one system in the Indian Point Study, so no division of basic events into Low Pressure and High Pressure systems was made. The two significant recirculation failures (Figure 20) are failure of all three safety injection (SI) pumps (SIP3123233) and a β factor calculation for errors in design, installation, or manufacturing which effect the SI pumps or certain MOVs (SIPMOV888). These events are discussed in Section 1.6.2.3.4.4 of the Indian Point Study. The fire events are also significant relative to recirculation failures and operator error. The operator errors most significant are failure to switch plant systems into recirculation configuration. The failure to do this during medium or large LOCAs (FIS19) is the most significant operator error. The value of FIS19 on Table 42 is larger than that of FI513. This reflects the more stressful environment for large and medium LOCAs compared to small LOCAs. The operator errors are also described in Section 1.6.2.3.4.4 of the Indian Point Study.

3. Criterion 3, Total Plant Risk

The results for Criterion 3 are the same as for Criterion 1, due to the fire sequence's release probability of 1.0 in a high consequences category (2). The results are listed on Table 47 and 48 and shown on Figure 21.

Table 40

Indian Point 3

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Westinghouse	4	Dry	965	Independent

Dominant Accident Sequences

Sequence	Contribution to Overall Criteria Impact Measures		
	1	2	3
1 SEFIRE	.93	.32	.93
3 SLFHPRC	-	.43	-
4 LLLPRC	-	.06	-
5 MLLPRC	-	.06	-

Dominant Accident Sequence Release Category Probabilities

Sequence	Reactor Safety Study PWR Release Categories							Sequence Frequency
	1	2	3	4	5	6	7	$(\text{R-yr})^{-1}$
1 SLFHPRC								8.2E-5
2 SEFIRE		1.0						6.1E-5
3 LLLPRC								1.1E-5
4 MLLPRC								1.1E-5

Total Plant Core Melt Frequency 1.9E-4

Table 41

Indian Point 3
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
A	RSSMAP Large LOCA	LLOCA
FIRE	Fire Induced Transients	CSRTE SWGEAR
OPERROR	Operator Error	FIS135 FIS19 MOV746747 RCP3132 RP3132RHR SIPS 1802ABRHR
RCIR	Recirculation System	HDWE MOV888883 MSIP P60 RCP3132 RP3132RHR SIPMOV888 SIP3 SIP313233 1802ABRHR
S1	RSSMAP S1 LOCA	MLOCA
S2	RSSMAP Small LOCA	SLOCA

Table 42

Indian Point 3
Boolean Terms from Sequence Fault Trees

Boolean Term	*Description	Value
CSRTE	Fire in Tunnel Entrance of Cable Room	2.4E-5
FIS13	Operator Failure to Initiate Switch to Recirculation During Small LOCA	3.5E-4
FIS19	Operator Error to Switchover to Recirculation During Large or Medium LOCA	4.8E-3
HDWE	Factor Calculation to Account for Undetermined Unavailability of all SI Pumps and MOVs Given Large LOCA or Medium LOCA	3.3E-5
LLOCA	Large LOCA	2.0E-3/R-yr
MLOCA	Medium LOCA	2.0E-3/R-yr
MOV746747	Switch 5 and 6 are Erroneously Turned to "on" Position, No Recovery Action Taken	5.3E-4
MOV888883	MOVs 88A and 888B Do Not Open, and Pipe 190 is Not Used	5.9E-7
MSIP	Maintenance on One SI Pump	2.4E-9
P60	Pipe 60 Rupture	1.7E-8
RCP3132	Mechanical Failure in Both Recirculation Pumps and Operator Error in Switching to Residual Heat Removal System (RHR) Pumps	1.6E-6
RP3132RHR	Hardware Failure in Both Recirculation Pumps and Operator Error in Switching to RHR Pumps	3.7E-5

*Most Boolean term descriptions are taken from Tables 1.6.2.3.4-13 and 1.6.2.3.4-19 of the Indian Point Probabilistic Safety Study.

Table 42 (Cont.)

Indian Point 3
Boolean Terms from Sequence Fault Trees

Boolean Term	Description	Value
SIPMOV888	Factor Calculation to Account for Undetermined Unavailability of all SI Pumps and MOVs	5.3E-4
SIPS	Operator Error, Switch 6 is Turned to "on" position and No Recovery Actions Taken	4.7E-5
SIP3	Maintenance on one Safety Injection pump and mechanical failure of other two	1.4E-5
SIP313233	All Three SI Pumps Fail	3.1E-3
SLOCA	Small LOCA	2.0E-2/R-yr
SWGEAR	Fire in Switchgear Room	3.6E-5
1802ABRHR	Hardware Failure in Both MOVs 1802 and B, Operator Failure to Switch to RHR Pumps	1.3E-6

Table 43

Indian Point 3

Contribution of Front Line and Support Systems to Total Core Melt Frequency With Release: Criterion 1

(Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
FIRE	.92	1.0	1.0

Table 44

Indian Point 3

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency and Release: Criterion 1

(Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
FIRE	SWGEAR	.55	.63	.60
	CSRTE	.37	.45	.40

Table 45

Indian Point 3

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2 (Vesely-Fussell Importance Measure)

System, Oper[illegible] Action [illegible] Initiati[illegible]	Importance	Worst Case Importance	Proportional Importance
S_2	.42	.57	.50
RCIR	.39	.53	.45
FIRE	.32	.46	.37
OPERROR	.15	.30	.18
A	.11	.26	.13

Table 46

Indian Point 3

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criterion 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
S_2	SLOCA	.42	.57	.50
RCIR	SIP313233	.33	.48	.38
	SIPMOV888	.06	.20	.07
	SIP3	.001	.15	.002
	RP3132RHR	.0008	.15	.0009
	HOWE	.0007	.15	.0008
	RCP3132	.0002	.15	.0002
	MOV888883	6.2E-5	.15	7.2E-5
	1802ABRHR	2.6E-5	.15	3.1E-5
	P60	1.8E-6	.15	2.1E-6
	MSIP	4.6E-8	.15	5.4E-8
FIRE	SWGEAR	.19	.34	.22
	CSRTE	.13	.27	.15
OPERROR	FIS19	.10	.25	.11
	FIS13	.04	.18	.04
	MOV746747	.01	.15	.01
	SIPS	.005	.15	.006
	RP3132RHR	.0008	.15	.0009
	RCP3132	.0002	.15	.0002
	1802ABRHR	2.6E-5	.15	3.1E-5
A	LLOCA	.05	.20	.06
	MLOCA	.05	.20	.06

Table 47

Indian Point 3

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
FIRE	.91	1.0	1.0

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
FIRE	SWGEAR	.55	.63	.60
	CSRTE	.37	.45	.40

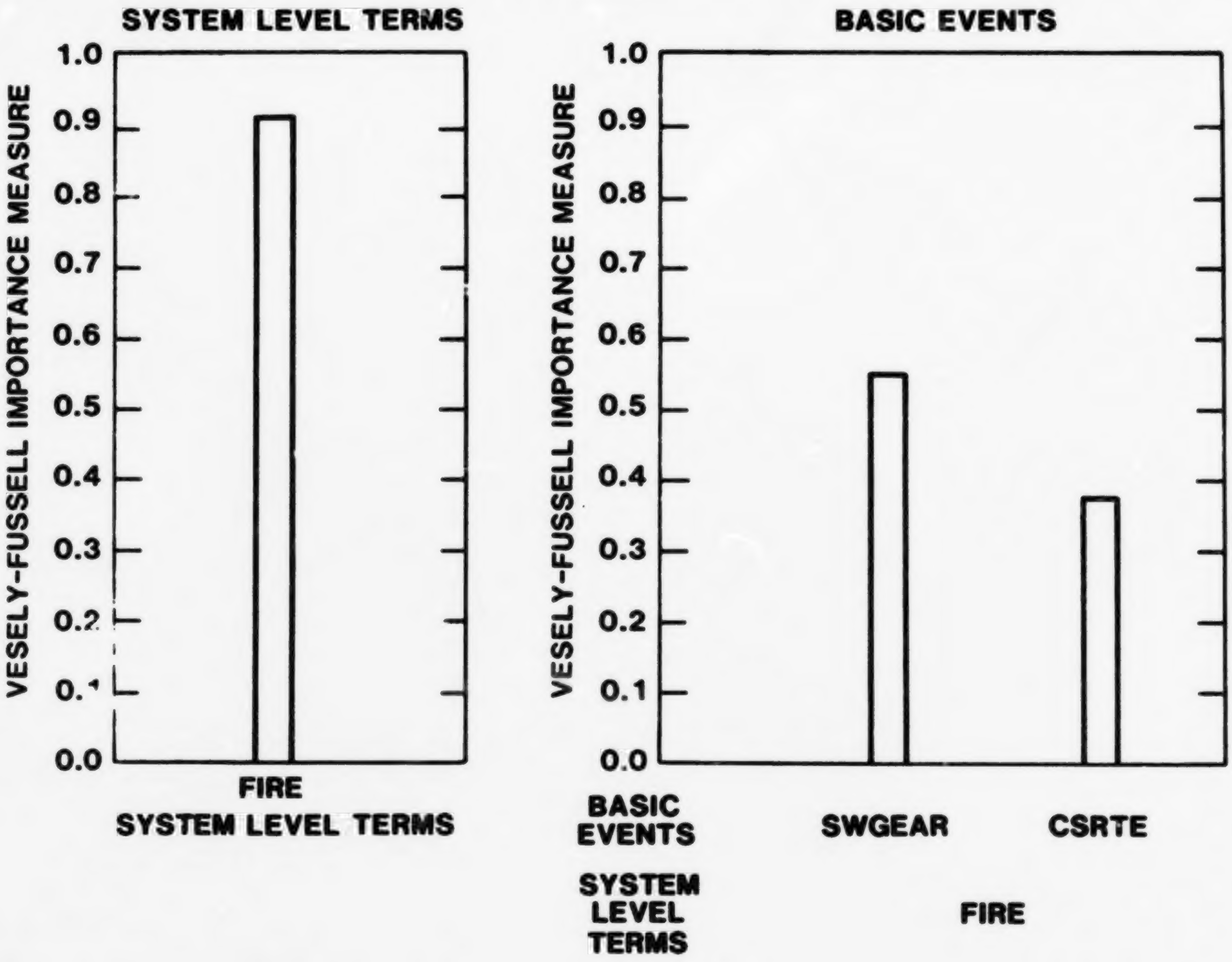

Figure 18. Indian Point 3, Importance of Front Line and Support Systems, and Basic Events to Total Core Melt Frequency with Release: Criterion 1

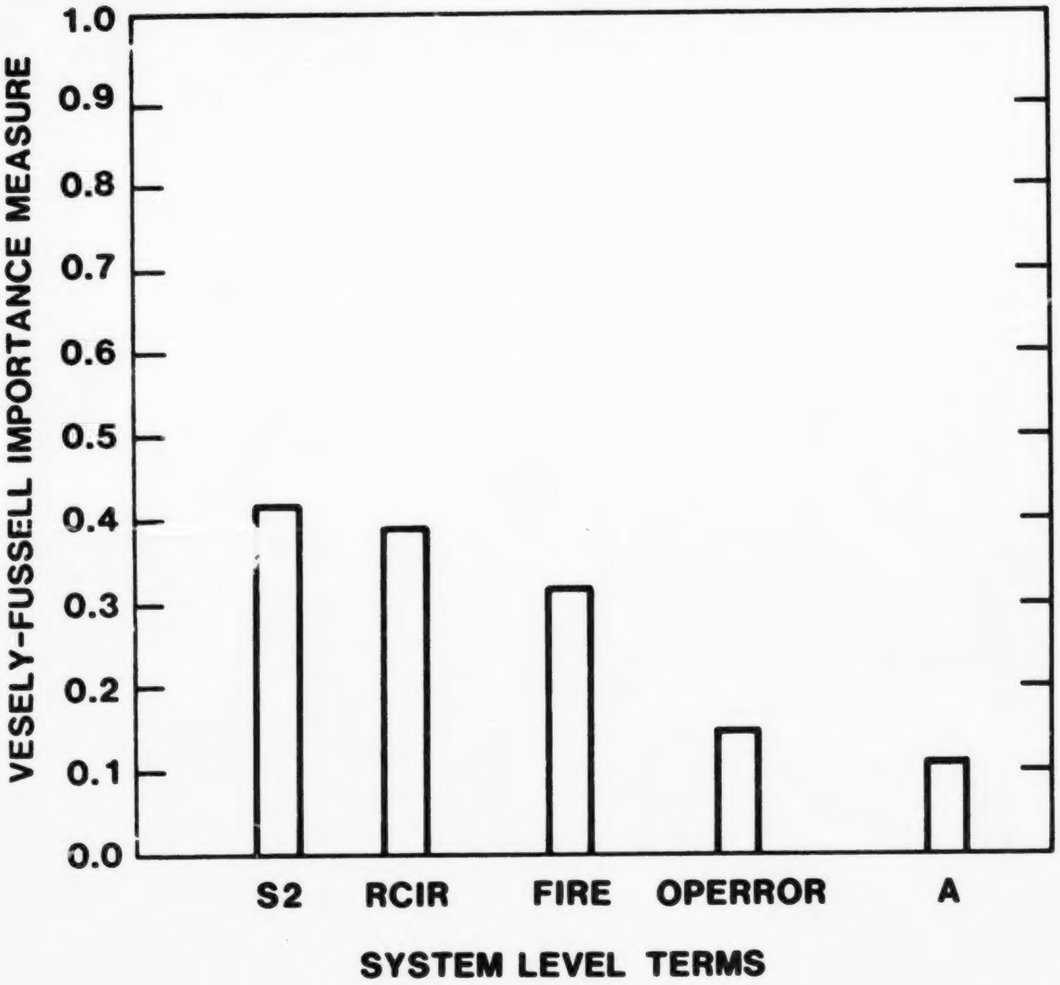

Figure 19. Indian Point 3, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2

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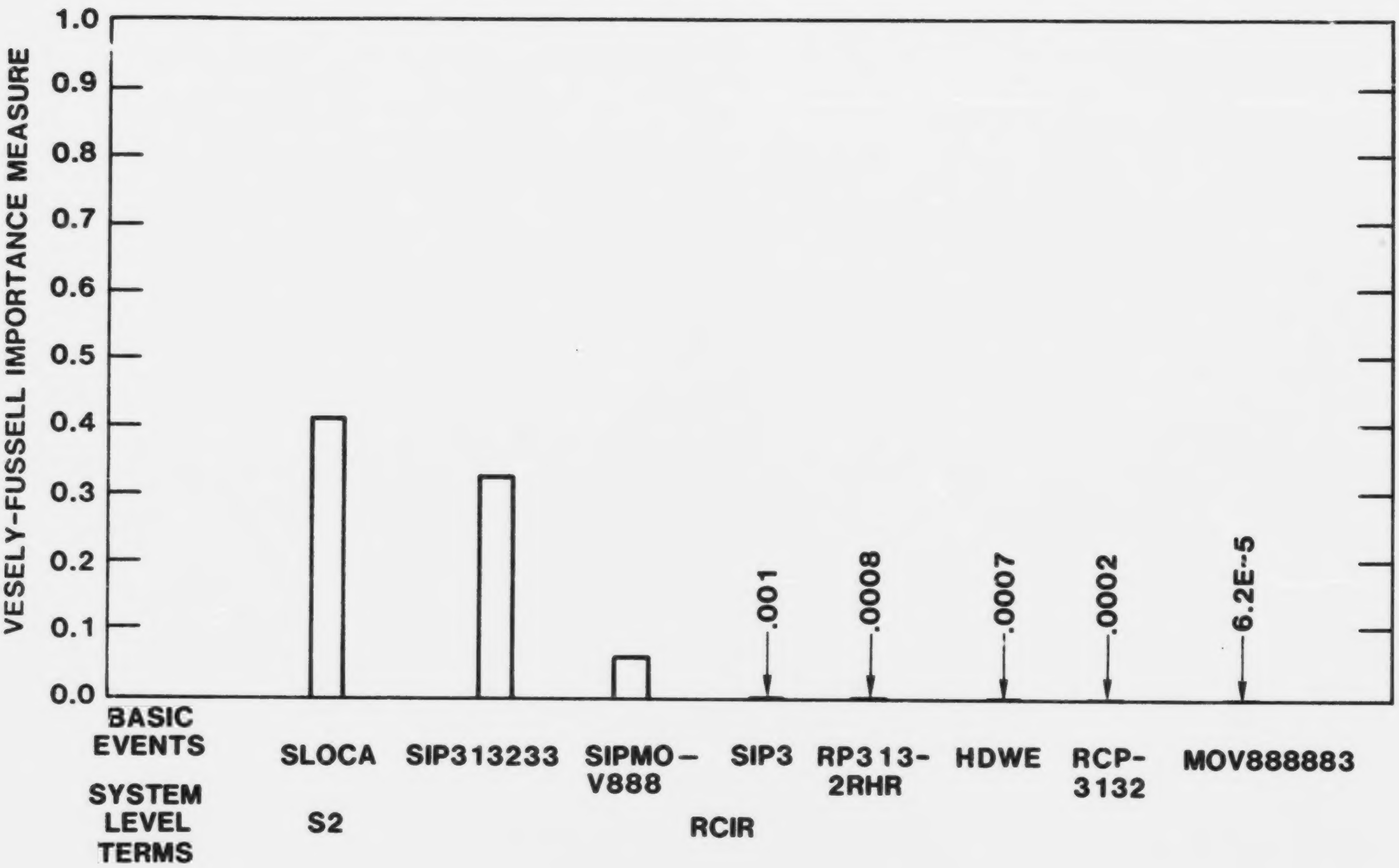

Figure 20. Indian Point 3, Importance of Basic Events Grouped by Systems, to Total Core Melt Frequency: Criterion 2

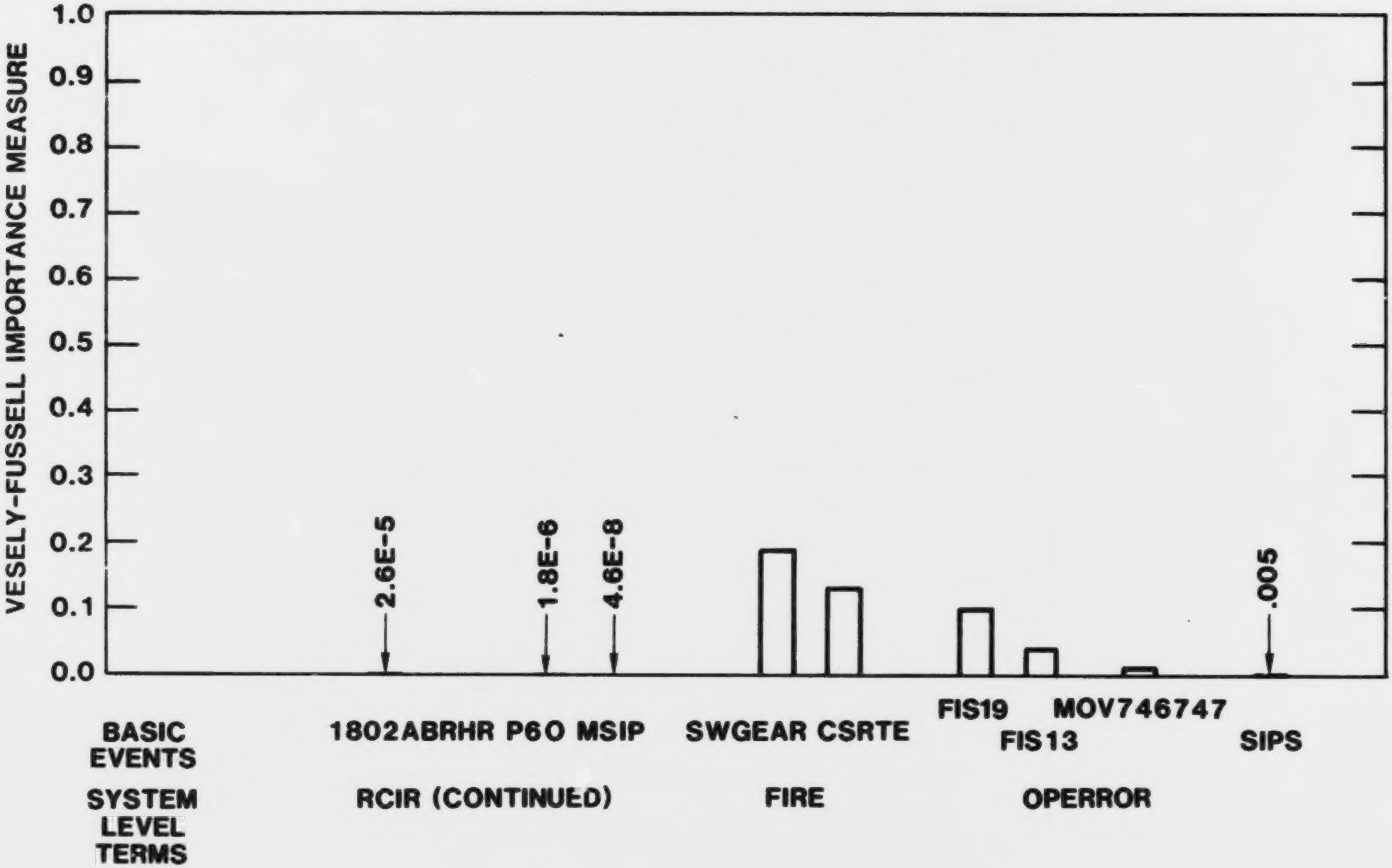

Figure 20. Indian Point 3, Importance of Basic Events Grouped by Systems, to Total Core Melt Frequency: Criterion 2 (Cont)

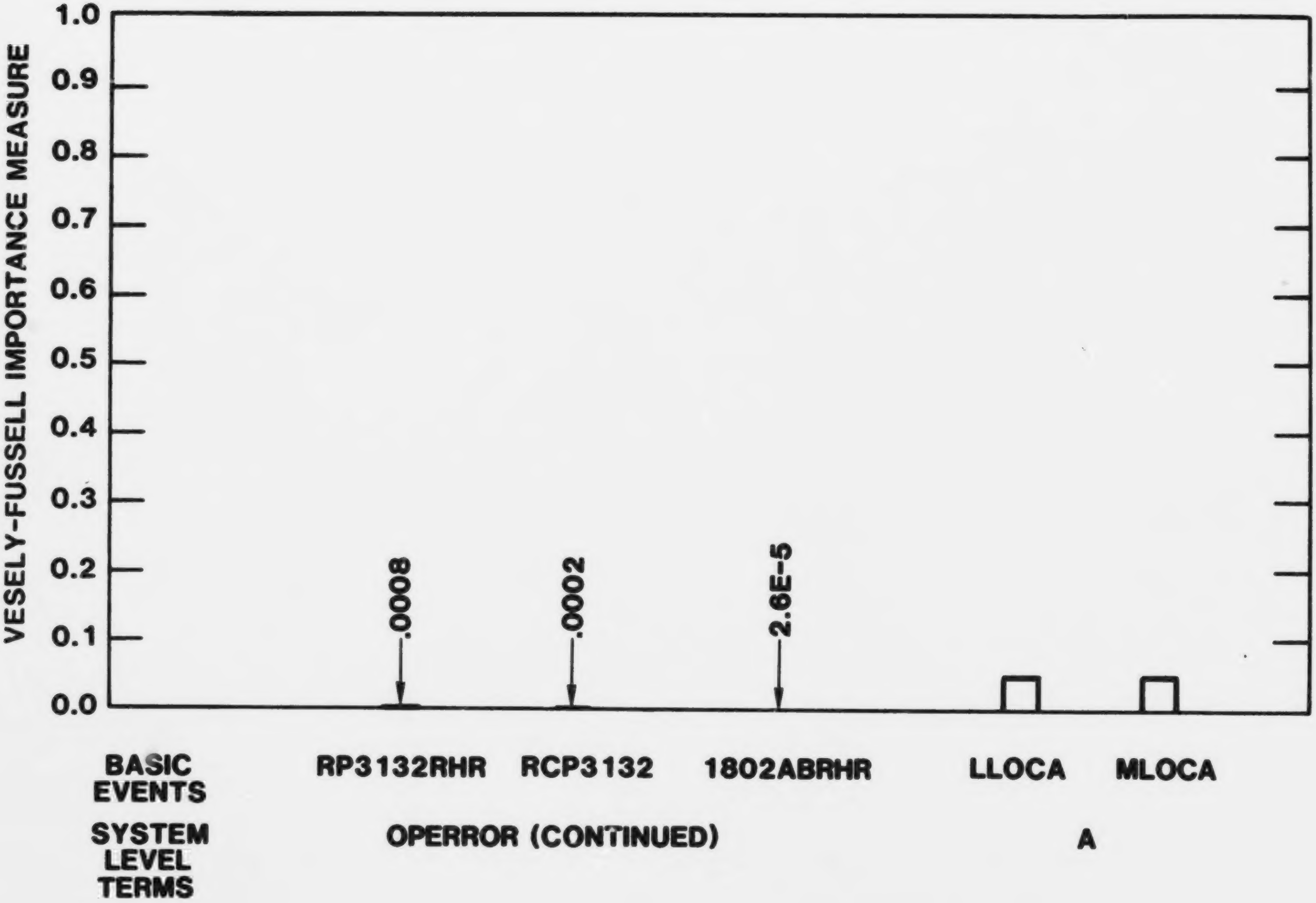

Figure 20. Indian Point 3, Importance of Basic Events Grouped by Systems, to Total Core Melt Frequency: Criterion 2 (Cont)

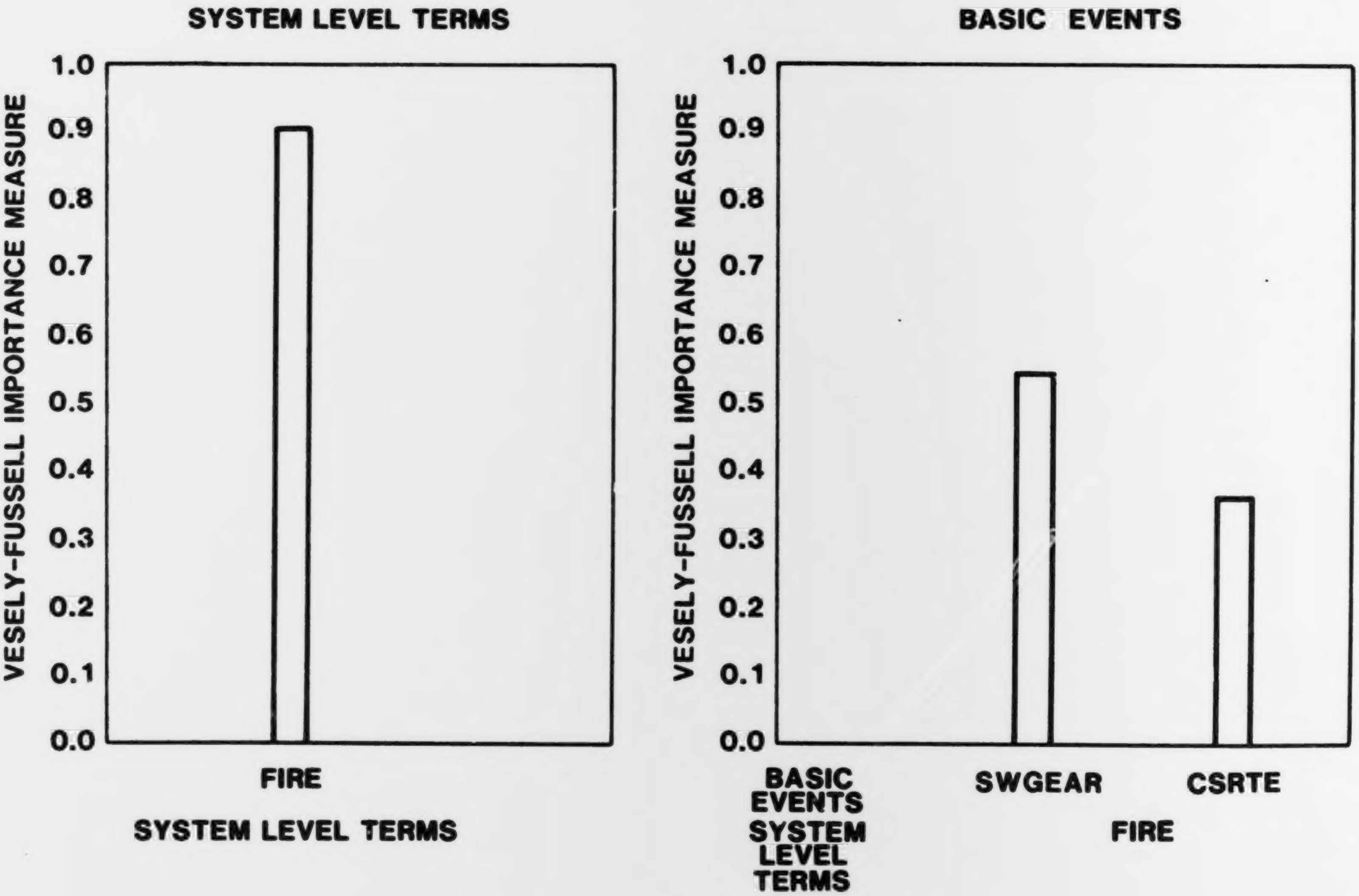

Figure 21. Indian Point 3, Importance of Front Line and Support Systems, and Basic Events to Total Plant Risk: Criterion 3

2. Babcock and Wilcox Plants

Crystal River-3
ANO-1
Oconee

a. Crystal River-3

The analysis for Crystal River was done using the Crystal River-3 Safety Study.[13] This report is the first Interim Reliability Evaluation Program (IREP) PRA. It was prepared for the NRC by Science Applications, Inc. Section 2.0 of the PRA contains descriptions of the dominant core melt sequences. For each sequence, the most dominant cut sets are listed, and the basic events of each cut set are described. The cut sets for this analysis were taken directly from that section, as were the basic event definitions.

1. Criteria 1 and 2, Total Core Melt Frequency

Table 48 shows only three sequences dominant to the core melt frequency for Crystal River. The cut sets for these sequences model about 73 percent of the total core melt frequency. This is attributed to the limited cut set information published in the PRA, and the exclusion of less dominant sequences from the analysis. However, given the uncertainties of data and modeling in PRAs, 73 percent of the core melt frequency is sufficient to gain an understanding of the relative importances of various systems and components to core melt frequency.

The system level importances (Table 51 and Figure 22) show that small-small LOCAs as defined in RSSMAP (S_2) are, by far, the most important initiating event. S_2 type LOCAs are basic event B_4 in the PRA. Operator errors are clearly the most dominant type of fault. The DC and emergency AC power systems have significant contributions as well. The importance of the Emergency Feedwater System is approaching the level of insignificance.

The basic event results are listed on Table 49 and shown on Figure 23. The three dominant operator errors involve improper operator actions during switchover to recirculation cooling mode or during recirculation mode. H*02 is improper alignment of valves for switchover from injection to recirculation. L06 is premature shutdown of the low pressure pumps while they are still needed during recirculation. L017 is premature switchover to recirculation. H*02 is described under HPRS failures in Appendix G of the PRA. L017 is under LPIS faults, and L06 is under LPRS faults of Appendix K-1.

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Table 48

Crystal River 3

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Babcock and Wilcox	2	Dry	906	IREP

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures	
	1	2	3
1 B_4S_{23}	.18	.18	.23
2 B_4S_2	.46	.46	.42
3 $T_{2A}T_{10}$	.15	.15	.18

Dominant Accident Sequence Release Category Probabilities

Sequence			Reactor Safety Study PWR Release Categories					Sequence Frequency $(R\text{-}yr)^{-1}$
	1	2	3	4	5	6	7	
1 B_4S_{23}	.0001	0.5		.007		0.5		6.6E-5
2 B_4S_2	.0001		0.5		.007		.05	1.7E-4
3 $T_{2A}T_{10}$	.0001	.05		.007		0.5		5.4E-5

Total Plant Core Melt Frequency 3.7E-4

Table 49

Crystal River
Steam Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
A	Large LOCA/Medium LOCA	B1 B2
DCPOWER	DC Power System	DCB
DHCCS	Decay Heat Closed Cycle Cooling System	DM2
EACPOWER	Emergency AC Power System	A2 A3 A5 AM1
EFWS	Emergency Feedwater System	EM2 E01 E1 E2
OPERROR	Operator Error	H4 H01 H03 H*02 L04 L06 L017
S1	Small LOCA, RSSMAP S1 LOCA	B3
S2	Small-Small LOCA, RSSMAP S2 LOCA	B4
T1	Loss of Offsite Power	T2A
T2	Transient; PCS Not Available, Offsite Power Available	T2-T2A

Table 50

Crystal River 3
Boolean Terms from Sequence Fault Trees

Boolean Term	Description	Value
AM1	Diesel Train A Down Due to Test or Maintenance	1.6E-2
A2	Failure of Backup AC Power from Crystal River 1 and 2	0.36
AM1	Diesel Train A Unavailable Due to Test or Maintenance	1.6E-2
A3	Failure of Diesel A to Start and Run	6.1E-2
A5	Failure of Diesel B to Start and Run	6.2E-2
B1	LARGE LOCA-Breach of Reactor Coolant System (RCS) with Flow Area Larger Than One Square Foot	1E-4/R-yr
B2	MEDIUM LOCA-Breach of RCS with Flow Area Between One Square Foot and .04 Square Feet	1E-4/R-yr
B3	SMALL LOCA-Breach of RCS with Flow Area Between .4 Square Feet and .087 Square Feet	1E-4/R-yr
B4	Small-Small LOCA	1.3E-3/R-yr
DCB	Failure of Battery B	3.2E-3
DM2	Train B Maintenance Outage in Decay Heat Removal Closed Cycle Coding System	2.1E-3
D2	Hardware Fault in Train B of Decay Closed Cycle Cooling System	2.0E-3
EM2	Maintenance and Test Outages in Motor Driven Emergency Feedwater Pump Train	5.5E-3
EO1	Auto Actuation of Emergency Feedwater is Locked Out and Operator Fails to Recover	1E-4

Table 50 (Cont.)

Crystal River 3
Boolean Terms from Sequence Fault Trees

Boolean Term	Description	Value
E1	Failure of Turbine Driven Emergency Feedwater Train	2.2E-2
E2	Hardware Faults in Motor Driven Emergency Feedwater Pump	3.5E-3
H01	Premature Cutoff of High Pressure Cooling Injection by Operator Given B4, B3, or Transient Induced LOCA	4E-3
H*02	Operator Error in Realigning Valves to Switch from High Pressure Injection to Recirculation	0.08
H03	Operator Failure to Diagnose Need for Feed and Bleed	1.4E-2
H4	One of Two Manual Valves in High Pressure Injection Train B is Left Open	2.0E-2
L04	Operator Fails to Initiate Recirculation Given a LOCA in Time to Avoid Draining Borated Water Storage Tank	3E-3
L06	Operator Error by Shutting Down Low Pressure Pumps Too Soon During Recirculation	0.06
L017	Operator Error in Switching to Recirculation Mode Too Soon, Causing Pump Failure	0.05
T2A	Loss of Offsite Power Initiating event	01.32/R-yr
T2-T2A	Transients with Power Conversion System Not Available, But With Offsite Power Available	1.78/R-yr

Table 51

Crystal River 3

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
S_2	.62	.90	.86
OPERROR	.62	.90	.86
T_1	.10	.38	.14
EACPOWER	.10	.38	.14
DCPOWER	.08	.35	.11
EFWS	.03	.30	.04

Table 52

Crystal River 3

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
S_2	B4	.62	.90	.86
OPERROR	H*02	.27	.55	.37
	L017	.17	.45	.24
	L06	.17	.45	.24
T_1	T_{2A}	.10	.38	.14
EACPOWER	A2	.10	.38	.14
	A3	.09	.37	.12
	A5	.03	.31	.04
	AM1	.02	.30	.02
DCPOWER	DCB	.08	.36	.11
EFWS	E1	.03	.30	.04

Table 53

Crystal River 3

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
S_2	.63	.87	.83
OPERROR	.63	.87	.83
T_1	.13	.37	.17
EACPOWER	.13	.37	.17
DCPOWER	.10	.33	.13
EFWS	.03	.27	.04

Table 54

Crystal River 3

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
S_2	S_2	.63	.87	.83
OPERROR	H*02	.25	.48	.32
	L017	.22	.46	.29
	L06	.16	.40	.21
T_1	T_{2A}	.13	.37	.17
EACPOWER	A2	.13	.37	.17
	A3	.11	.35	.15
	A5	.03	.27	.04
	AM1	.02	.26	.03
DCPOWER	DCB	.10	.33	.13
EFWS	E1	.03	.27	.04

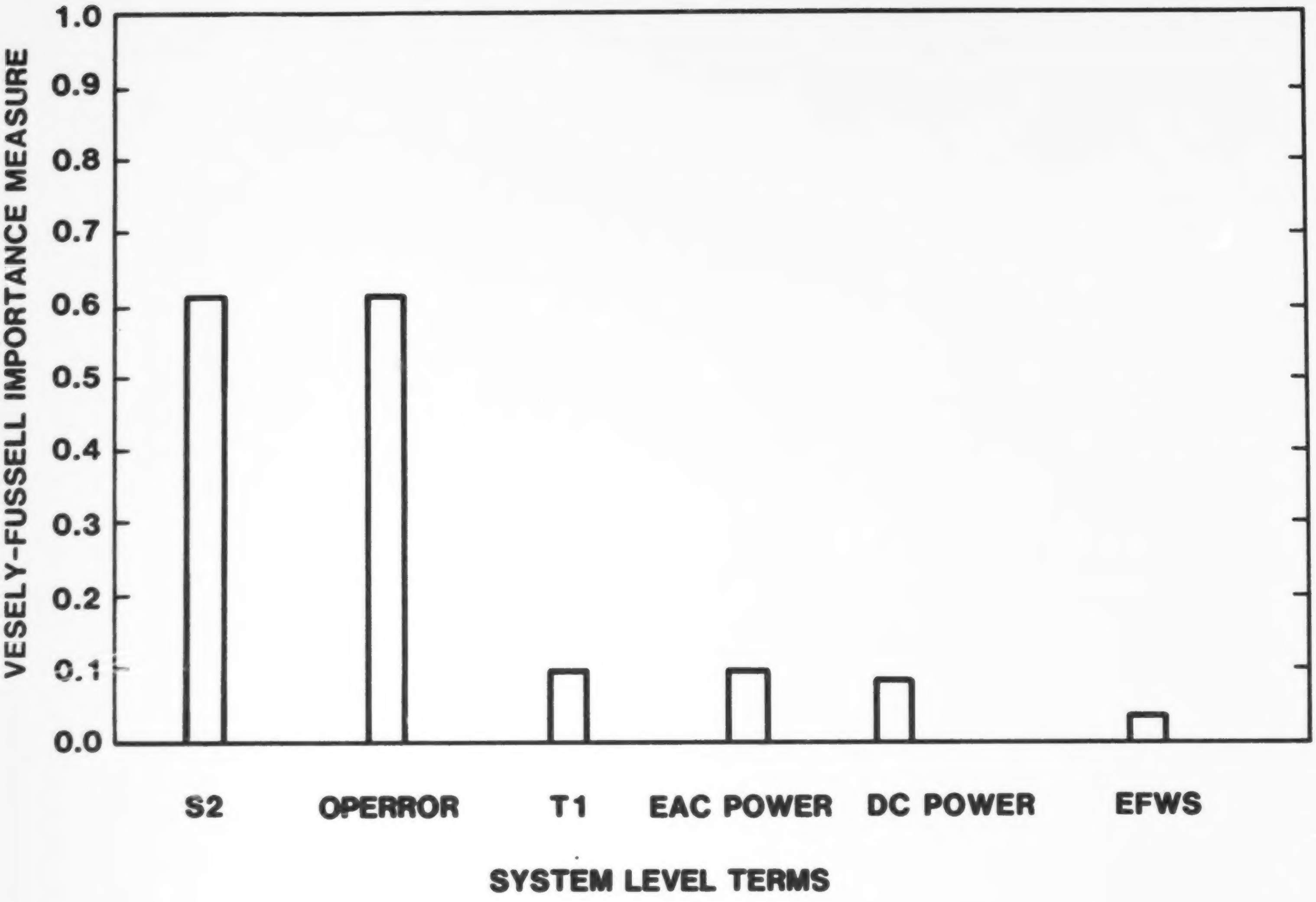

Figure 22. Crystal River 3, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

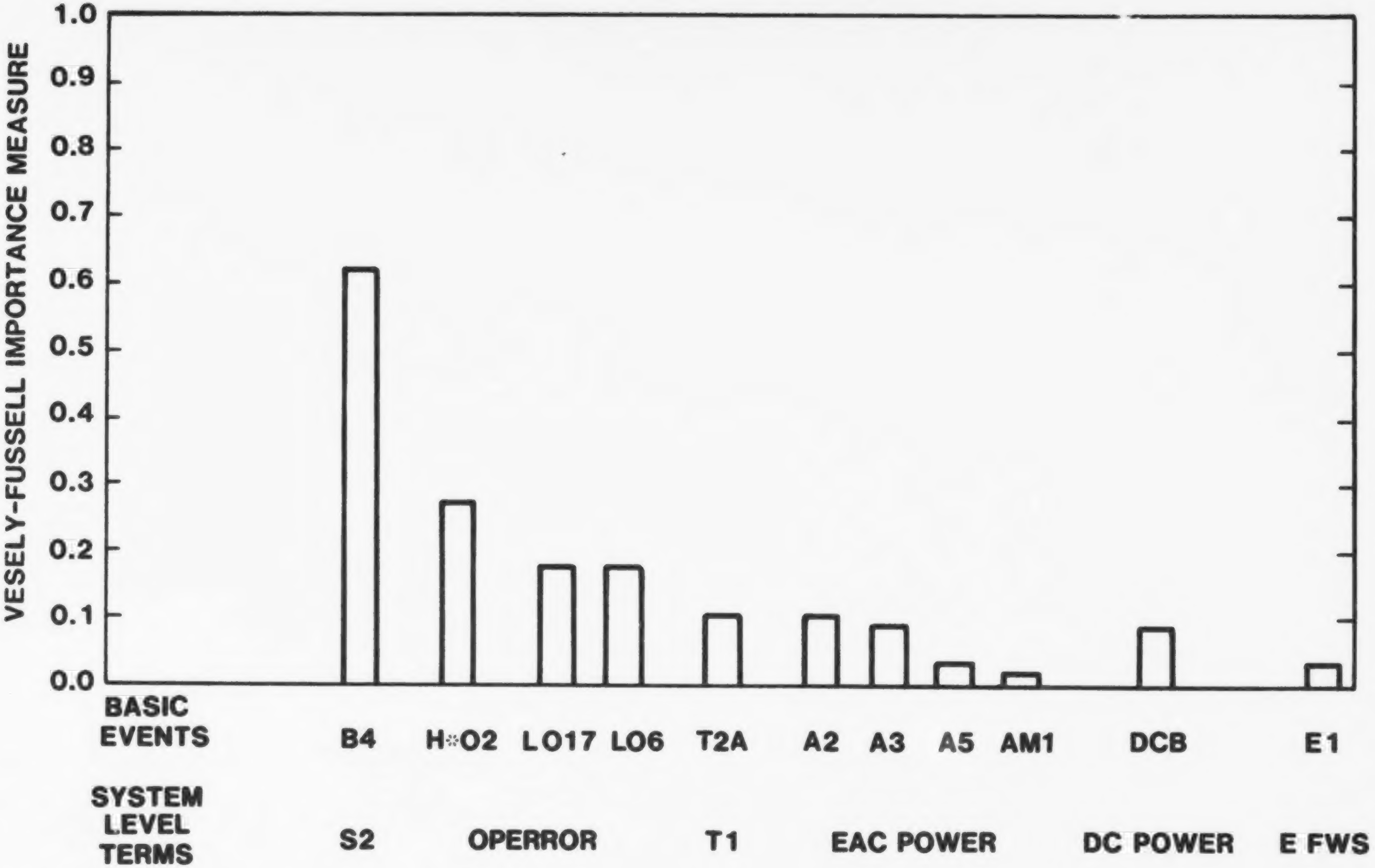

Figure 23. Crystal River 3, Importance of Basic Events Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

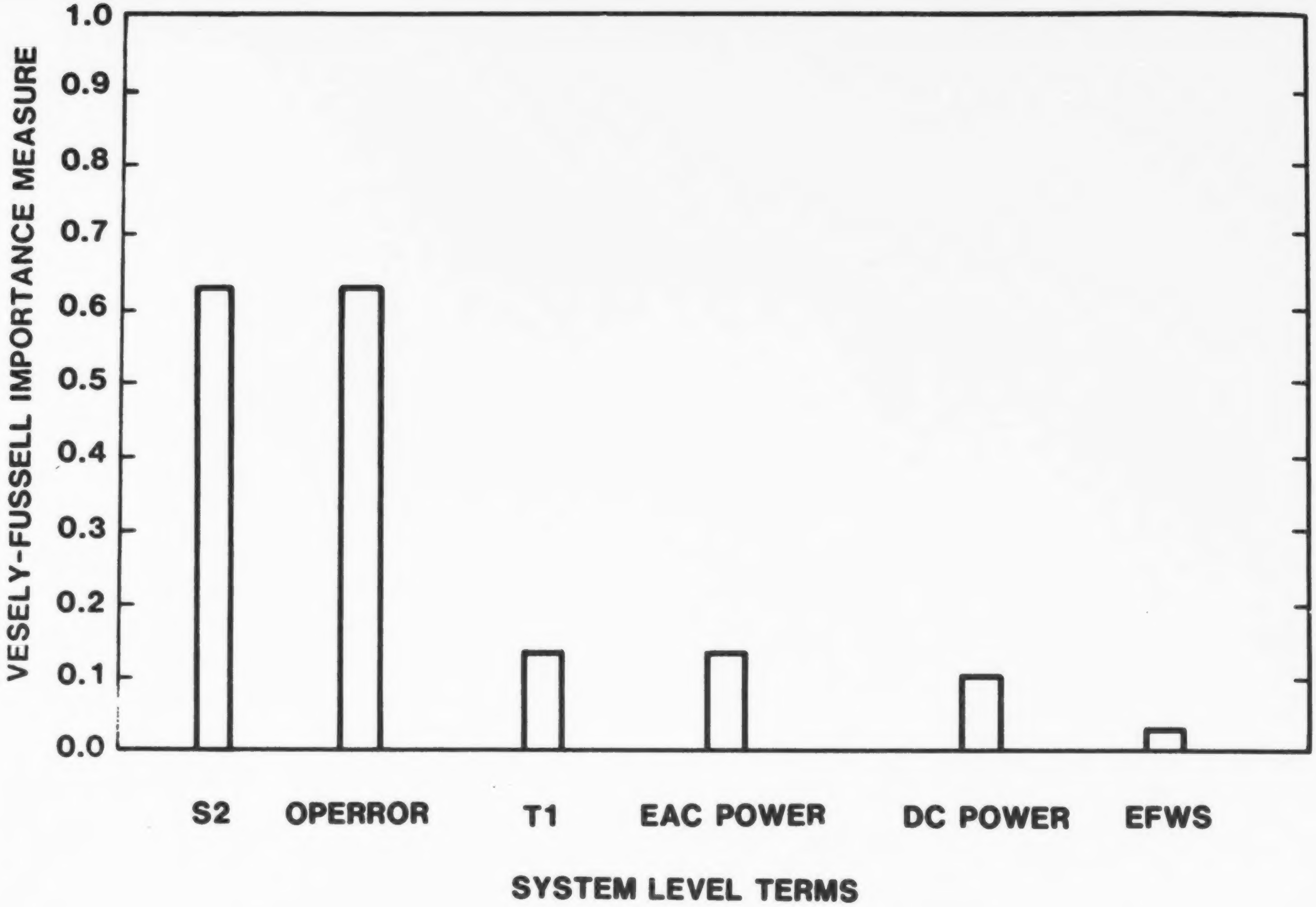

Figure 24. Crystal River 3, Importance of Front Line and Support Systems to Total Plant Risk: Criteria 3

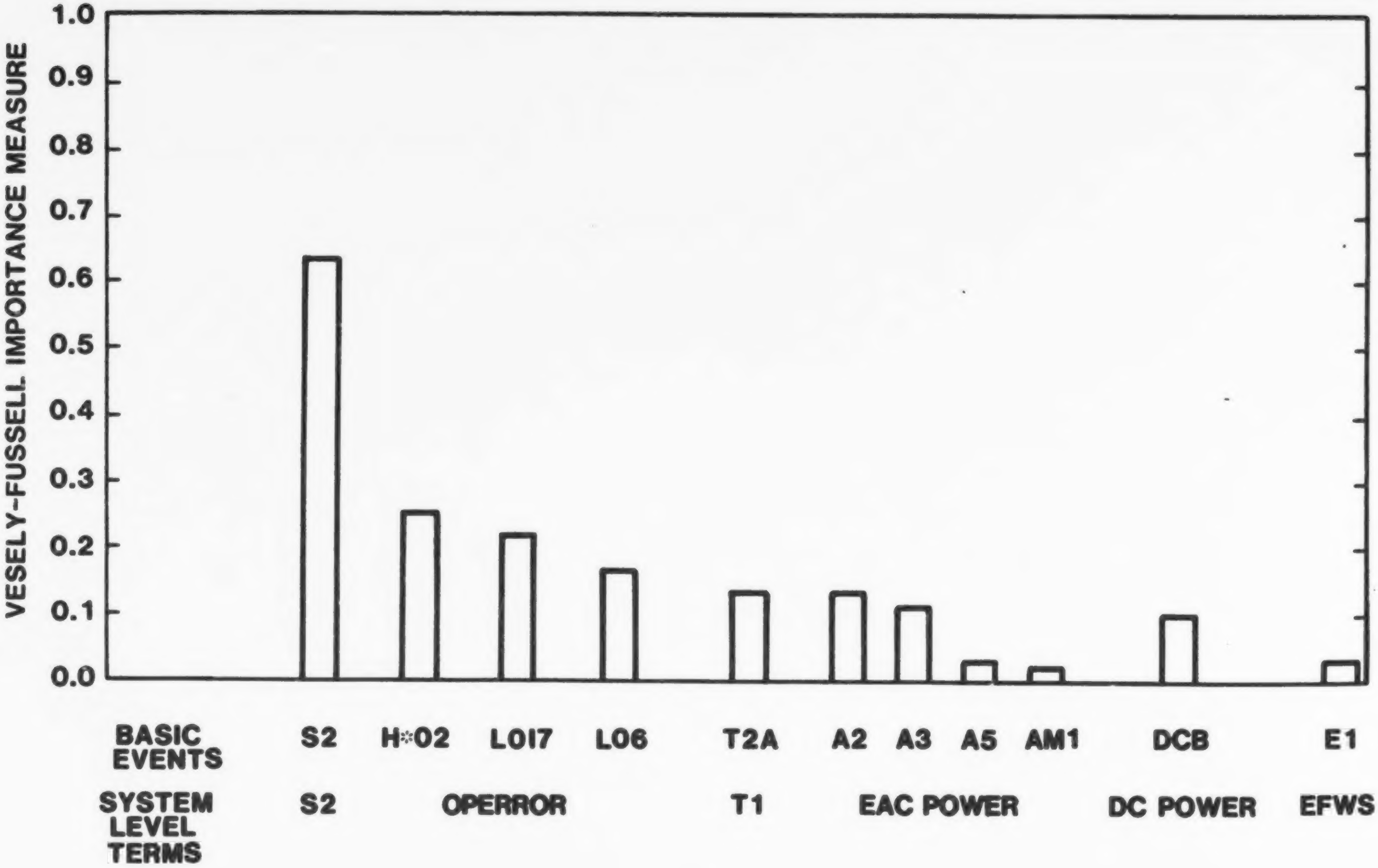

Figure 25. Crystal River 3, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

The emergency AC faults are described in Appendix D of the PRA. The individual unavailabilities of the two diesel generators (A3 and A5) are virtually equal. However, diesel train Bs (A5) lesser importance than train A (A3) appears to be due to train B's dependence on the B battery in the DC system. The breaker connecting diesel train B to the bus would not close with failure of the DC B train. Thus, A5 is excluded from any cut set involving failure of DC train B (basic event DCB).

The DC system is described in Appendix C of the PRA. The two DC supplies, A and B, have equivalent unavailabilities. However, train B is more important to core melt frequency. While failure of either train is considered sufficient to fail DC power, failure of DCB automatically fails diesel generator train B, removing one of Crystal River's three backup AC systems. The emergency feedwater system (EFWS) is described in Appendix D of the PRA. E1 is hardware failure in EFWS train A, predominantly due to failure of the turbine pump to start.

2. Criterion 3, Total Plant Risk

The results from Criterion 3 are very similar to those for Criteria 1 and 2. The same sequences dominate all three Criteria, with the risk weighted cut sets modeling 76 percent of the plant risk. The system level and basic event importances (Tables 53 and 54, Figures 24 and 25) are almost exactly the same as for Criteria 1 and 2.

b. Arkansas Nuclear One-1

The analysis for ANO-1 was done using the ANO-1 IREP PRA. Dominant cut sets for the dominant accident sequences were taken from Appendix C of the PRA and are included in Appendix A7 of this report. The cut sets analyzed from the PRA model 65 percent of total core melt frequency (Criteria 1 and 2) and only 59 percent of plant risk. Of the fourteen sequences discussed in the PRA, five sequences were eliminated by the 5 percent test for dominant sequences for this analysis. These account for 14 percent of the core melt frequency. The remaining gap between core melt frequency and plant risk model and actual core melt frequency and plant risk is due to limited inclusion of sequence and cut set information in the PRA.

1. Criteria 1 and 2, Total Core Melt Frequency

Eight sequences dominate core melt frequencies (Table 55). All the ANO-1 sequences have frequencies fairly close in value to each other. Thus, it is not surprising that many system level terms have similar importance measures (Table 58, Figure 26). DC power is the most important plant system, but seven other plant systems and operator actions have relatively significant contributions. Inspection of the basic event results (Table 59, Figure 27) shows that each system level term has only one to three significant basic events.

DC power is dominated by the single most dominant basic event (EBAT-CM). This is a common mode failure caused by human error during test and maintenance which allows the batteries to degrade. This fault is discussed in Appendix D of the PRA. The other DC faults involved battery or bus failures and are described in Appendix A12 of the PRA.

Basic event Q is also an event tree level event on ANO-1 transient event trees. The event, failure of pressurizer relief valves to reseat, is discussed in Appendix A of the PRA. Event Q appears in each cut set of the two dominant sequences that it appears in (T(A3)LQ-D3, and T(D01)LQ-D3). In the event of loss of EFWS (Event 2 of the sequences), the SRVs are automatically demanded.

The EFWS is dominated by three faults, all described as local failures in pipe segments. The events are described in Appendix B5 of the PRA. Event EFW-E8 is a check valve failure, while EFW-E4 and EFW-E29 have several subevents. The cut sets for the various dominant sequences (Appendix A7) show that cut sets with EFW-E8 apparently are less dependent on other EFWS faults than cut sets containing other EFWS terms. This appears to be the source of event EFW-E8's dominance of EFWS.

Both dominant operator errors involve failure to initiate High Pressure Injection Systems (HPIS). Both events were named HPI-PUMP-CM in the PRA, but were renamed PUMP for transient sequences and A2 for LOCA sequences to differentiate between the high stress environment of the RPS fails (PUMP), and the less stressful environment of LOCAs (A2). The human error events are discussed in Appendix B15.

The HPIS and Reactor Building Spray Injection System (RBSI) share two basic events, A4 (called LPI1408B-VCC in the PRA) and LPI-L25 are pipe segments or valve faults with fail suction to HPIS pumps and RBSI pump B. These faults are discussed in Appendix A2 of the PRA. HPIS faults A1(LPI1407A-VCC) and HPI-H14 fail HPIS pumps A and B, and pump C, respectively. A1 is also discussed in Appendix B.2 of the PRA. HPI-H14 is discussed in Appendix B.1 of the PRA.

The Reactor Protection System (RPS) is discussed in Appendix B.7 of the PRA. Trip breakers B and A (events A8 and A11) dominant the RPS importance.

Table 55

ANO-1

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Babcock and Wilcox	2	Dry	820	IREP

Dominant Accident Sequences

Sequence	Contribution to Overall Criteria Impact Measures		
	1	2	3
1 $B(1.2)D_1$	.06	.06	.07
2 $B(1.2)D_1C$	.09	.09	.12
3 $T(A3)LQ-D_3$	.07	.07	.08
4 $T(D01)LQ-D_3$	.08	.08	.10
5 $T(LOP)LD_1YC$	.21	.21	.11
6 $T(D01)LD_1YC$	.06	.06	–
7 $T(D02)LD_1YC$	.05	.05	–
8 $T(FIA)KD_1$	.06	.06	.07
9 $T(D01)LD_2$	–	–	.06

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study PWR Release Categories						Sequence Frequency
	1	2	3	4	5	6	7	$(R\text{-}yr)^{-1}$
1 $B(1.2)D_1$	.0001	0.5			.007		0.5	2.8E-6
2 $B(1.2)D_1C$	.0001	0.5		.007		0.5		4.4E-6
3 $T(A3)LQ-D_3$	.0001	0.5			.007		0.5	3.3E-6
4 $T(D01)LQ-D_3$	.0001	0.5			.007		0.5	4.0E-6
5 $T(LOP)LD_1YC$	.0001	0.2		.007		0.8		9.9E-6
6 $T(D01)LD_1YC$	.0001	0.2		.007		0.8		3.1E-6
7 $T(D02)LD_1YC$	.0001	0.2		.007		0.8		2.5E-6
8 $T(FIA)KD_1$	.0001	0.5			.007		0.5	2.8E-6
9 $T(D01)LD_1$	.0001	0.5			.007		0.5	2.2E-6

Total Plant Core Melt Frequency 4.8E-5

Table 56

ANO-1
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
ACPOWER	Onsite AC Power System	AC-A3 AC-A4 AC-B5 AC-B6
DCPOWER	DC Power System	DC-D01 DC-D02 DC-D06 DC-D07 EBAT-CM
EACPOWER	Emergency AC Power System	AC-DG1 AC-DG2
ECS	Emergency Cooling System	A3 ECS-ROOM99 SWS-VCH4A SWS-VCH4B
EFIC	Emergency Feedwater Initiation and Control System	EFC-ABCD4 EFC-ACCD4 EFC-BB7B1 EFC-CSY2 EFC-D1D2 EFC-VCD1 EFC-VDC2 EFC-VE2 EFC-VE4
EFWS	Emergency Feedwater System	EFW-E2 EFW-E3 EFW-E4 EFW-E5 EFW-E8 EFW-E22 EFW-E28 EFW-E92
HPIS	High Pressure Injection System	A2 A4 HPI-H14 LPI-L25

Table 56 (Cont.)

ANO-1
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	DESCRIPTION	COMPONENT
HPPRC	High Pressure Pump Room Cooling System	HPI-ROOM55
LPIS	Low Pressure Injection System	A1
LPRS	Low Pressure Recirculation System	A6 A7
OPERROR	Operator Error	A2 HPI-PUMP HPRS-CM PUMP
Q	Pressurizer Safety/Relief Valve Fails to Close	Q
RBSI	Reactor Building Safety Injection System	A4 A5 LPI-L25
RPS	Reactor Protection System	A8 A9 A10 A11 A12 A13
SWS	Service Water System	SWS-ROOM99 SWS-1 SWS-2 SWS-5 SWS-15 SWS-62 SWS-63 SWS-82 SWS-83
S3	Small LOCA	B(1.2) B(1.66) B(4)
T(A3)	Transient Initiated by Failure of Engineered Safeguards (ES) Bus A3	T(A3)

Table 56 (Cont.)

ANO-1
System-Level Grouping of Boolean Terms From Sequence Fault Trees

SYSTEM	DESCRIPTION	COMPONENT
T(D01)	Transient Initiated by Failure of ES Bus D01	T(D01)
T(D02)	Transient Initiated by Failure of ES Bus D02	T(D02)
T1	Transient, Loss of Offsite Power	TLOP
T3	Transients with All Front Line Systems Initially Available	T(FIA)

Table 57

ANO-1
Boolean Terms from Sequence Fault Trees

Boolean Term	Descriptions	Value
*AC-A3	Local Fault of Engineering Safeguards (ES) Bus 43 fails High Pressure Injection System (HPIS) Pumps A and B, Reactor Building Cooling System (RBCS) Fans A and B, Reactor Building Safety Injection System (RBSI) Pump A, Emergency Feedwater System (EFS) Electrical Pump)	2.4E-4
AC-44	Local Fault in ES Bus A4	2.4E-4
AC-B5	Local Fault in ES Bus B5 (480VAC) (Fails HPIS Pumps A and B Suction, RBCS Fans A and B, RBSI Pump A Suction, 1/2 EFS Electric Pump Flow Control Valves)	4.4E-4
AC-B6	Local Fault in ES Bus B6	4.4E-4
AC-DG1	Local Fault of Diesel Generator 1 (Fails HPIS A and B Pumps, RBCS A and B Fans, RBSI A Pump, EFS Electric Pump)	3.3E-2
AC-DG2	Local Fault of Diesel Generator 2 (Fails HPIS C Pump, RBCS C and D Fans, RBSI B Pump)	3.3E-2
**A1=LPI1407A-VCC	Local Faults of Low Pressure Injection System (LPIS) Valve CV1407 Fails A and B Pumps Suction of HPIS	8.2E-3
A2=HPI-PUMP-CM	Failure of Operator to Initiate HPIS	1E-4

*AC-A3 appears in the PRA as LF-AC-A3. For simplicity, all LFs that either preceded or followed a Boolean term were dropped.

**Boolean terms A1 through A13 were introduced and equated to PRA terms due to computer code field limitations that only allowed for ten alphanumeric characters.

Table 57 (Contd.)

Boolean Term	Descriptions	Value
A3=ECS-ROOM100	Local Faults in AC Room Cooler 100 ("odd" AC Power Fails Causing SWS Pump which Feeds the HPRS Room Heat Exchanger to Fail)	5E-3
A4=LPI1408B-VCC	Local Fault of LPIS Valve CV1408 (Fails Suction to HPIS C Pump and RBSI B Pump)	8.2E-3
A5=LPI1408B-VCC	Local Fault of LPIS Valve CV1408 (Fails Suction to HPIS C Pump and RBSI B Pump)	8.2E-3
A5=(LF-RBI-B1+ LF-RBI-B9)	Local Faults in Pipe Segments B1/BS	3.4E-2
A6=(LF-LPI-L17+ LF-LPI-L19)	Local Faults in LP Pipe Segment (L19 and L210 Fail Suction to all High Pressure Recirculation System (HPRS)	2.6E-2
A7=(LF-LPI-L20+ LF-LPI-L18)	Local Faults in LP Pipe Segments L20/L18	2.6E-2
A8=RPS000BB-BCC	Reactor Trip Breaker B Fails to Open (B and D1 or D2 Will Cause RPS Failure	1E-3
A9=RPS00D2D-BCC	Reactor Trip Breaker D2 Fails to Open (D2 and B Will Cause RPS Failure)	1E-3
A10=RPS00D1D-BCC	Reactor Trip Breaker D1 Fails to Open (D1 and B Will Cause RPS Failure)	1E-3
A11=RPS000AA-BCC	Reactor Trip Breaker A Fails to Open (A and C, or C2 Will Cause RSP Failure)	1E-3
A12=RPS00C2C-BCC	Reactor Trip Breaker C2 Fails to Open (C2 and A Will Cause RSS Failure)	1E-3

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Table 57 (Cont.)

Boolean Term	Descriptions	Value
A13=RPS00C1C-BCC	Reactor Trip Breaker C1 Fails to Open (C1 and A Will Cause RPS Failure)	1E-3
B(1.2)	Reactor Coolant Pump Seal Failure	2E-2/R-yr
B(1.66)	Small LOCA (1.2" D 1.66")	3.1E-4/R-yr
B(4)	Reactor Cooling System (RCS) Pipe Break 1.66" D 4"	3.8E-4/R-yr
DC-D01	Local Fault of ES Bus D01 (125VDC) (Fails HPIS Pumps A and B, RBCS Fans A and B, RBSI Pump A, 1/2 EFS Turbine Pump Flow Control Valves, and EFS Electric Pump)	1E-4
DC-D02	Local Fault of ES Bus D02 (125VDC) (Fails 1/2 EFS Turbine Pump Flow Control Valves, HPIS C Pump, RBCS Fans C and D, RBSI Pump B)	1E-4
DC-D06	Local Fault of Battery D06 (Fails HPIS A and B Pumps, EFS Electric Pump and 1/2 Turbine Pump Flow Control Valves, RBCS A and B Fans, RBSI A Pump)	1.1E-3
EBAT-CM	Common Mode Failure of Both Station Batteries (Fails all Mitigating Systems)	2.6E-5
ECS-ROOM99	Local Faults in AC Room Cooler 99 ("Even" AC Power Fails Causing Failure of HPRS Room Fan)	5E-3
EFC-ABCD4	Local Fault in Emergency Feedwater Initiation and Control System (EFIC) Initiate Signal Path	1.1E-2
EFG-ACBD4	Local Fault in EFIC Initiate Signal Path (Fails 1/2 EFS Turbine Pump Steam Admission Valves)	1.1E-2
EFC-BB7B1	Local Fault in EFIC Initiate Signal Path (Fails 1/2 EFS Turbine Pump Steam Admission Valves)	5.4E-3

Table 57 (Cont.)

Boolean Term	Descriptions	Value
EFC-CSY2	Local Fault in EFIC Initiate Signal Path (Fails 1/2 Turbine Pump Steam Admission Valves)	3.9E-3
EFC-D1D2CM	Local Fault in EFIC Vector Signal Path (Fails EFS Turbine Pump Flow Control Valves)	4.6E-3
EFC-VCD1	Local Fault in EFIC Vector Signal Path	9.5E-3
EFC-VCD2	Local Fault in EFIC Vector Signal Path (Fails 1/2 EFS Turbine Pump Flow Control Valves)	9.4E-3
EFC-VE2	Local Fault in EFIC Initiate Signal Input to Vector System	5.1E-3
EFC-VE4	Local Fault in EFIC Initiate Signal Input to Vector System	5.1E-3
EFW-E2	Local Fault in EFS Pipe Segment E2 (Fails 1/2 Turbine Pump Flow Path)	1E-4
EFW-E3	Local Fault in EFS Pipe Segment E3	1.2E-2
EFW-E4	Local Fault in EFS Pipe Segment E4 (Fails 1/2 Turbine Pump Flow Control Valves)	1.2E-2
EFW-E5	Local Fault in EFS Pipe Segment E5 (Fails 1/2 Turbine Pump Flow Control Valves)	1.2E-2
EFW-E8	Local Fault in EFS Pipe Segment E8 (Also Includes E11, E16, E17, E25, E26, E43)	4.1E-3
EFW-E22	Local Fault in EFS Pipe Segment E22 (Fails Suction Path to Both EFS Pumps)	3E-4
EFW-E28	Local Fault in EFS Pipe Segment E28 (Fails 1/2 Turbine Pump Steam Admission Valves)	8.1E-3

Table 57 (Cont.)

Boolean Term	Descriptions	Value
EFW-E29	Local Fault in EFS Pipe Segment E29 (Fails 1/2 EFS Turbine Pump Steam Admission Valves)	8.1E-3
HPRS-CM	Failure of Operator to Initiate the HPRS	1.6E-3
HPI-H14	Local Faults in HPIS Pipe Segment H14 (Fails C Pump)	1.4E-2
HPI-PUMP	Only Applies to Transients. It Implies Failure to Establish Feed-water Bleed	1E-4
HPI-ROOM55	Local Faults in HPRS Room Cooler	7.5E-4
LPI-L25	Local Faults of LPIS Segment L25 (Fails Pump Suction to All HPIS and RBSI Pumps)	1E-4
PUMP	Operator Fails to Initiate HPIS After Failure of RPS (High Stress)	0.1
Q	Failure of One of Two Pressurizer Safety/Relief Valves to Close After Being Demanded Open	2E-2
SWS-ROOM99	Failure to Establish Service Water to Room 99	5E-3
SWS-S1	Local Faults in Service Water System (SWS) Pipe Segment S1 (Fails AC and DC Room Cooler Which Fails HPRS Pump Room Fan AC/DC Power)	5E-3
SWS-S2	Local Faults in SWS Pipe Segment S2 (Fails A and B Pump Cooling)	5E-3
SWS-S5	Local Faults in SWS Pipe Segment S5 (Fails A and B Pump Cooling)	1E-2
SWS-S14	Local Faults in SWS Pipe Segment S14 (Fails A and B Pump Cooling)	1E-2
SWS-S62	Local Faults in Pipe Segment S62	8.6E-4

Table 57 (Cont.)

Boolean Term	Descriptions	Value
SWS-S63	Local Faults in Pipe Segment 63	8.6E-4
SWS-S82	Local Faults in SWS Pipe Segment S82 (Fails Cooling to HPRS Pump Room Heat Exchanger)	5E-3
SWS-S83	Local Faults in SWS Pipe Segment S83 (S83 and S82 Fail Suction Cooling to all HPRS)	2.3E-2
SWS-VCH4A	Local Faults in AC and DC Switchgear Room Cooler VCH4A (Fails HPIS Pump C and RBSI Pump B AC/DC Power Cooling	2.3E-2
SWS-VCH4B	Local Faults of AC and DC Switchgear Room Cooler VCH4B (Fails A and B Pumps AC/DC Power Cooling)	2.3E-2
T(A3)	Transient Initiated by Failure of the ES Bus A3 (4160VAC)	3.5E-2/R-yr
T(D01)	Transient Initiated by Failure of the ES Bus D01 (125VDC)	1.8E-2/R-yr
T(D02)	Transient Initiated by Failure of the ES Bus D02 (125VDC)	1.8E-2/R-yr
T(FIA)	Transients With all Front Line Systems Initially Available	7.1/R-yr
T(LOP)	Loss of Offsite Power Transient	3.2E-1/R-yr

Table 58

ANO-1

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
DCPOWER	.25	.61	.39
T_1	.19	.55	.30
Q	.15	.51	.24
T(D01)	.15	.51	.23
EFWS	.14	.50	.22
S_3	.14	.50	.22
OPERROR	.10	.46	.16
HPIS	.10	.46	.15
RBSI	.08	.44	.13
T(A3)	.07	.43	.10
RPS	.06	.42	.09
T_3	.06	.42	.09
EFIC	.05	.41	.08
T(D02)	.04	.40	.06
SWS	.02	.38	.04
ECS	.02	.37	.02
EACPOWER	.02	.37	.02
ACPOWER	.004	.36	.007

Table 59

ANO-1

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
DCPOWER	EBAT-CM	.17	.53	.27
	DC-D02	.04	.40	.06
	DC-D02	.04	.40	.06
	DC-D06	.003	.36	.005
	DC-D07	.003	.36	.005
T_1	TLOP	.19	.55	.20
Q	Q	.15	.51	.24
T(D01)	T(D01)	.15	.51	.23
EFWS	EFW-E8	.10	.46	.16
	EFW-E4	.03	.39	.04
	EFW-E29	.02	.38	.02
	EFW-E22	.007	.37	.01
	EFW-E5	.001	.36	.002
	EFW-E2	.008	.36	.001
	EFW-E28	.0005	.36	.0007
	EFW-E3	2.4E-5	.36	3.7E-5
S_3	B(1.2)	.14	.50	.22
OPERROR	PUMP	.06	.42	.09
	A2	.04	.41	.07
HPIS	A1	.05	.41	.07
	LPI-L25	.04	.41	.07
	A4	.03	.39	.05
	HPI-H14	.02	.37	.02
RBSI	A1	.05	.41	.07
	LPI-L25	.04	.41	.07
	A4	.03	.39	.05
	A5	4.5E-5	.36	7.1E-5
T(A3)	T(A3)	.07	.43	.10

Table 59 (Cont.)

ANO-1

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2
(Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
RPS	A8	.03	.39	.05
	A11	.03	.39	.05
	A9	.01	.38	.02
	A10	.01	.38	.02
	A12	.01	.38	.02
	A13	.01	.38	.02
T_3	T(FIA)	.06	.42	.09
EFIC	EFC-BB781	.01	.37	.02
	EFC-121D2	.009	.37	.01
	EFC-ACBD4	.007	.37	.01
	EFC-VCD2	.006	.37	.009
	EFC-CSY2	.003	.36	.004
	EFC-ABCD4	.0003	.36	.0004
	EFC-VCDZ	.0002	.36	.0003
	EFC-VE4	3.0E-5	.36	4.8E-5
	EFC-VE2	9.6E-6	.36	1 5E-5
T(D02)	T(D02)	.04	.40	.06
SWS	SWS-S1	.01	.38	.02
	SWS-S2	.002	.37	.004
	SWS-S62	.001	.37	.002
	SWS-S63	.001	.37	.002
	SWS-S14	.0009	.37	.001
	SWS-S5	.0009	.37	.001
	SWS-ROOM99	.0002	.37	.0003
EACPOWER	AC-DG2	.01	.37	.02
	AC-DG2	.01	.37	.02
ECS	ECS-ROOM99	.01	.37	.02
	SWS-VCH4A	.006	.37	.01
	SWS-VCH4B	.002	.37	.003
	A3	.0005	.37	.0007

Table 59 (Cont.)

ANO-1

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
ACPOWER	AC-B6	.002	.37	.003
	AC-B5	.001	.37	.002
	AC-A3	.0008	.37	.001
	AC-A4	.0005	.37	.0008

Table 60

ANO-1

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
EFWS	.19	.61	.33
Q	.19	.60	.32
S_3	.17	.59	.30
HPIS	.17	.58	.29
T(D02)	.15	.57	.26
OPERROR	.13	.54	.22
RBSI	.10	.52	.17
T_1	.10	.52	.17
DCPOWER	.09	.51	.16
T(A3)	.08	.50	.14
T_3	.07	.49	.13
RPS	.07	.49	.13
EFIC	.05	.47	.09
SWS	.01	.43	.03
EACPOWER	.008	.42	.01
ECS	.005	.42	.008
ACPOWER	.0005	.42	.0009

Table 61

ANO-1

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
EFWS	EFW-E8	.12	.54	.21
	EFW-E4	.05	.46	.08
	EFW-E29	.03	.45	.05
	EFW-E22	.009	.42	.15
	EFW-E2	.0009	.42	.002
	EFW-E5	.0008	.42	.001
Q	Q	.19	.60	.32
S_3	B(1.2)	.17	.59	.30
HPIS	HPI-H14	.07	.48	.11
	A1	.06	.48	.10
	LPI-L25	.05	.47	.09
	A4	.04	.46	.07
T(D01)	T(D01)	.15	.57	.26
OPERROR	PUMP	.07	.49	.13
	A2	.05	.47	.09
	HPI-PUMP	.002	.42	.004
RBSI	LPI-L25	.05	.47	.09
	A4	.04	.46	.07
T_1	TLOP	.10	.52	.17
DCPOWER	EBAT-CM	.09	.51	.16
	DC-D06	.002	.42	.003
	DC-D07	.002	.42	.003
T(A3)	T(A3)	.08	.50	.14
T_3	T(FIA)	.07	.49	.13

Table 61 (Cont.)

ANO-1

Contribution of Basic Events, Grouped by Front Line
and Support System, to total Plant Risk:
Criterion 3
(Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
RPS	A11	.04	.45	.06
	A8	.04	.45	.06
	A9	.02	.43	.03
	A10	.02	.43	.03
	A12	.02	.43	.03
	A13	.02	.43	.03
EFIC	EFC-D1D2	.009	.43	.02
	EFC-ACBD4	.009	.43	.02
	EFC-VCD2	.008	.42	.01
	EFC-BB7B1	.004	.42	.007
	EFC-CSY2	.003	.42	.005
	EFC-ABCD4	.0001	.42	.0002
	EFC-VCD1	.0001	.42	.0002
	EFC-VE2	1.2E-5	.42	2.0E-5
SWS	SWS-S1	.008	.42	.01
	SWS-S2	.003	.42	.005
	SWS-S5	.001	.42	.002
	SWS-S14	.001	.42	.002
	SWS-SG2	.0006	.42	.001
	SWS-SG2	.0006	.42	.001
EACPOWER	AC-DG1	.007	.42	.01
	AC-DG2	.007	.42	.01
ECS	SWS-VCH4B	.003	.42	.005
	SWS-VCH4A	.001	.42	.002
	A3	.0006	.42	.001
	ECS-ROOM99	.0002	.42	.0004
ACPOWER	AC-A3	.0004	.42	.0007
	AC-B5	.0002	.42	.0003

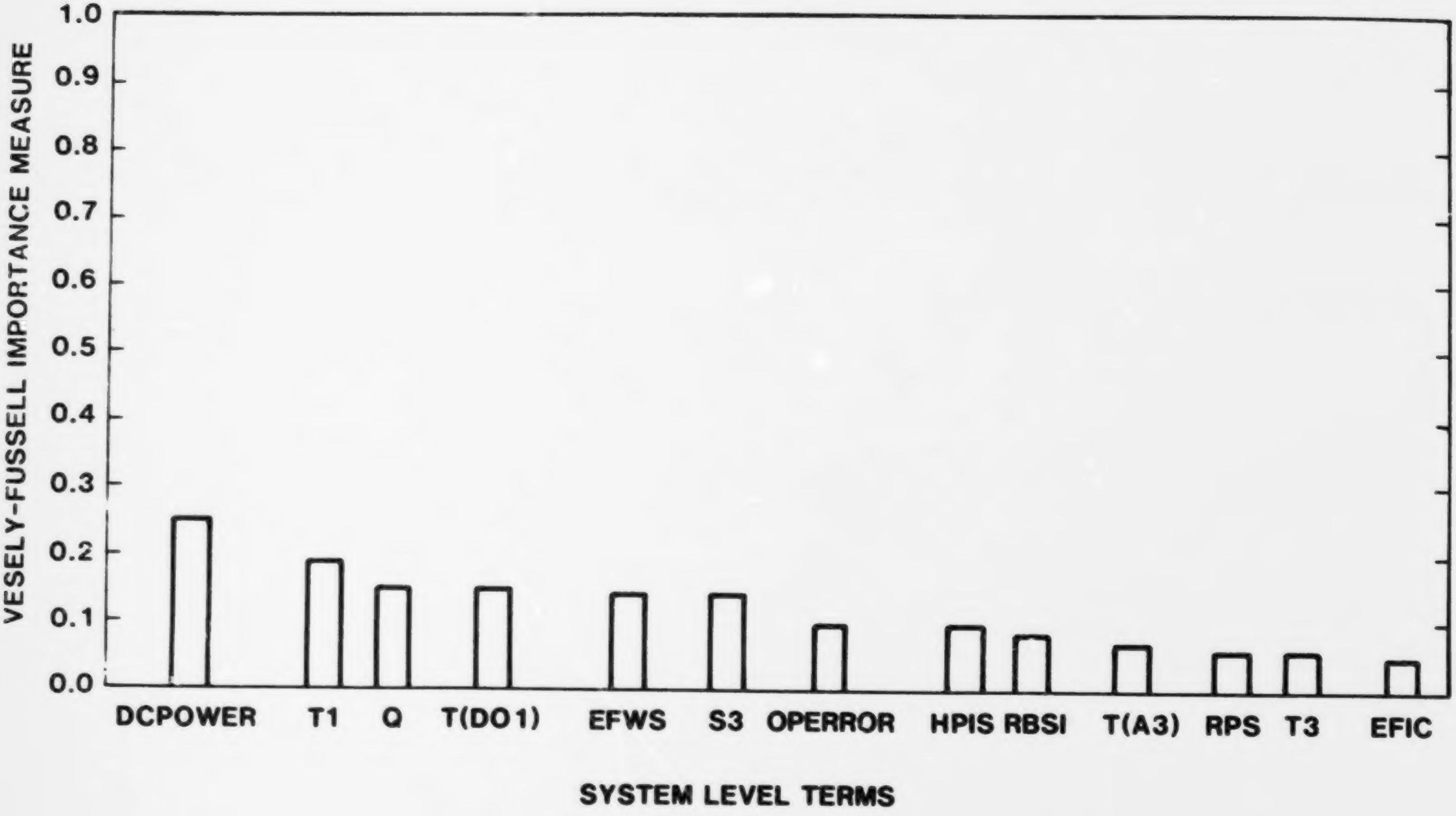

Figure 26. ANO-1, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

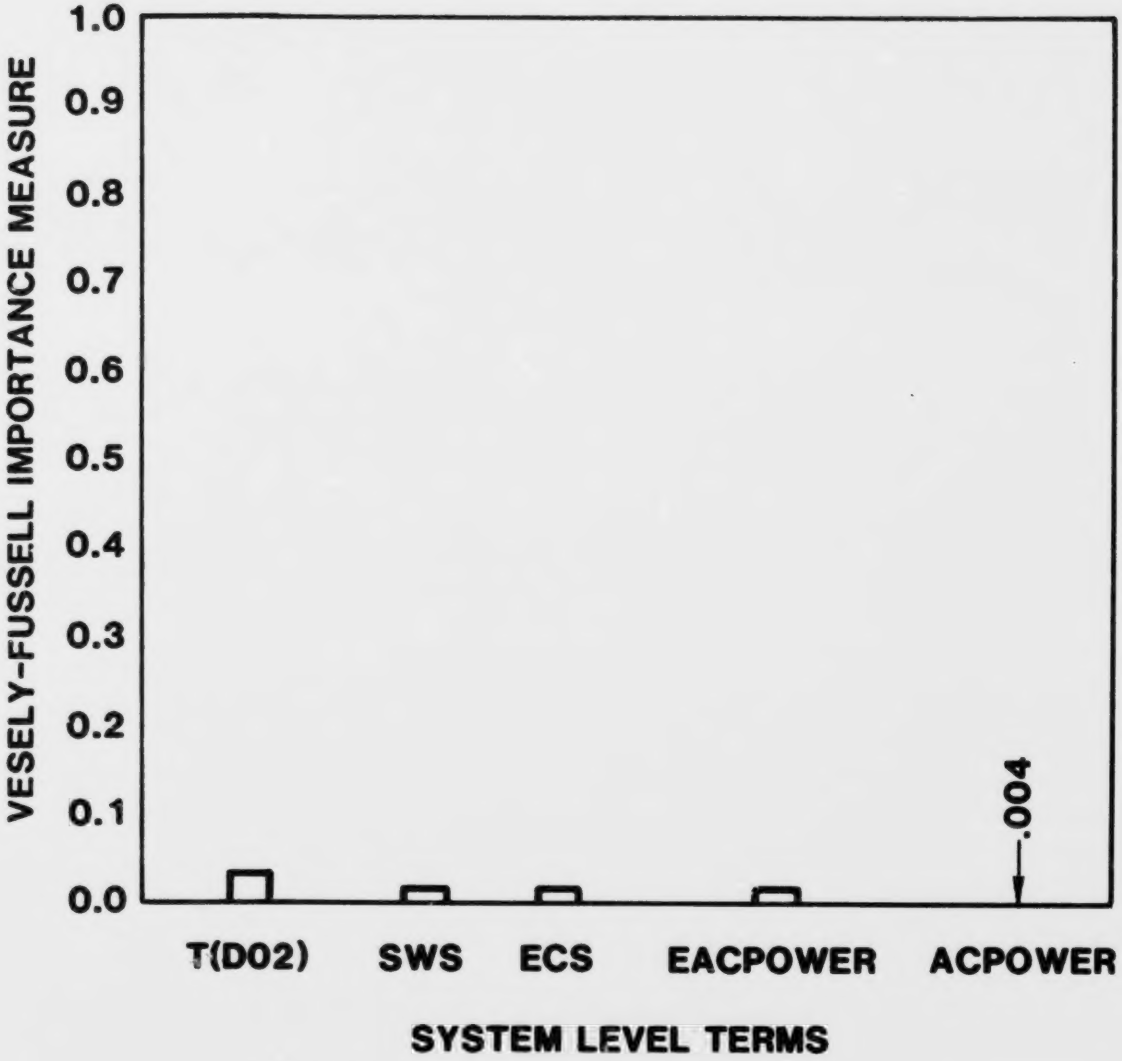

Figure 26. ANO-1, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Cont.)

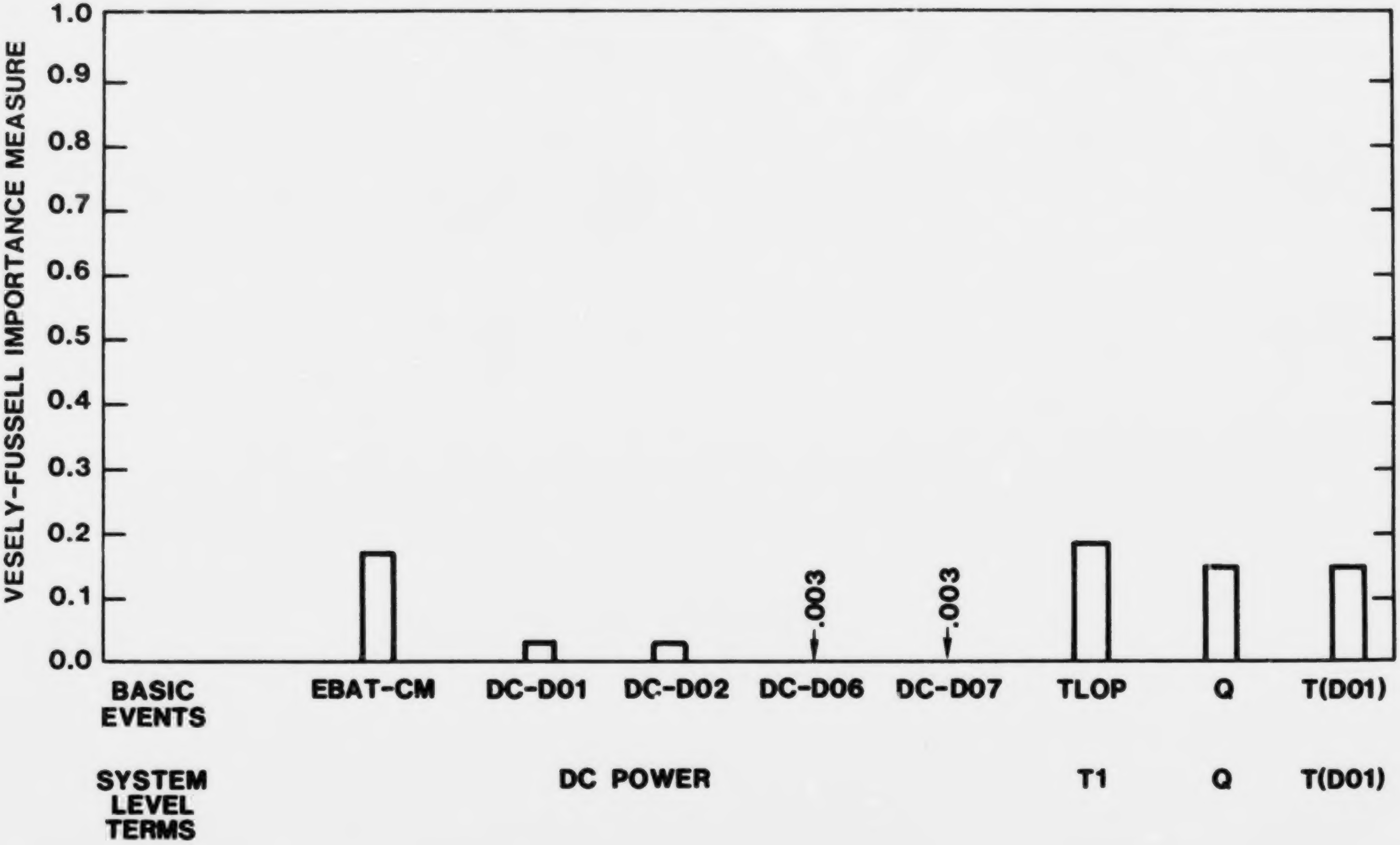

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

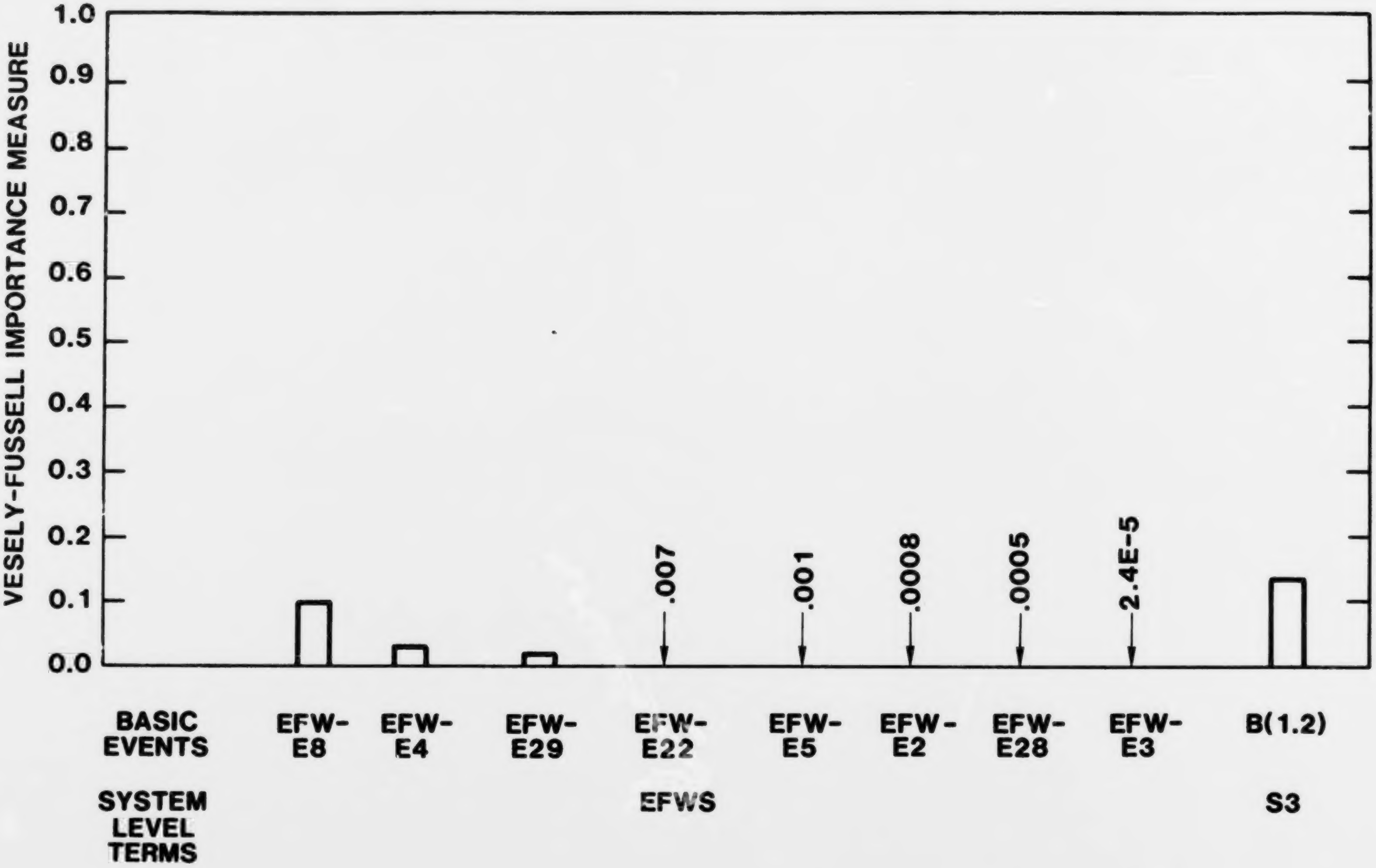

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

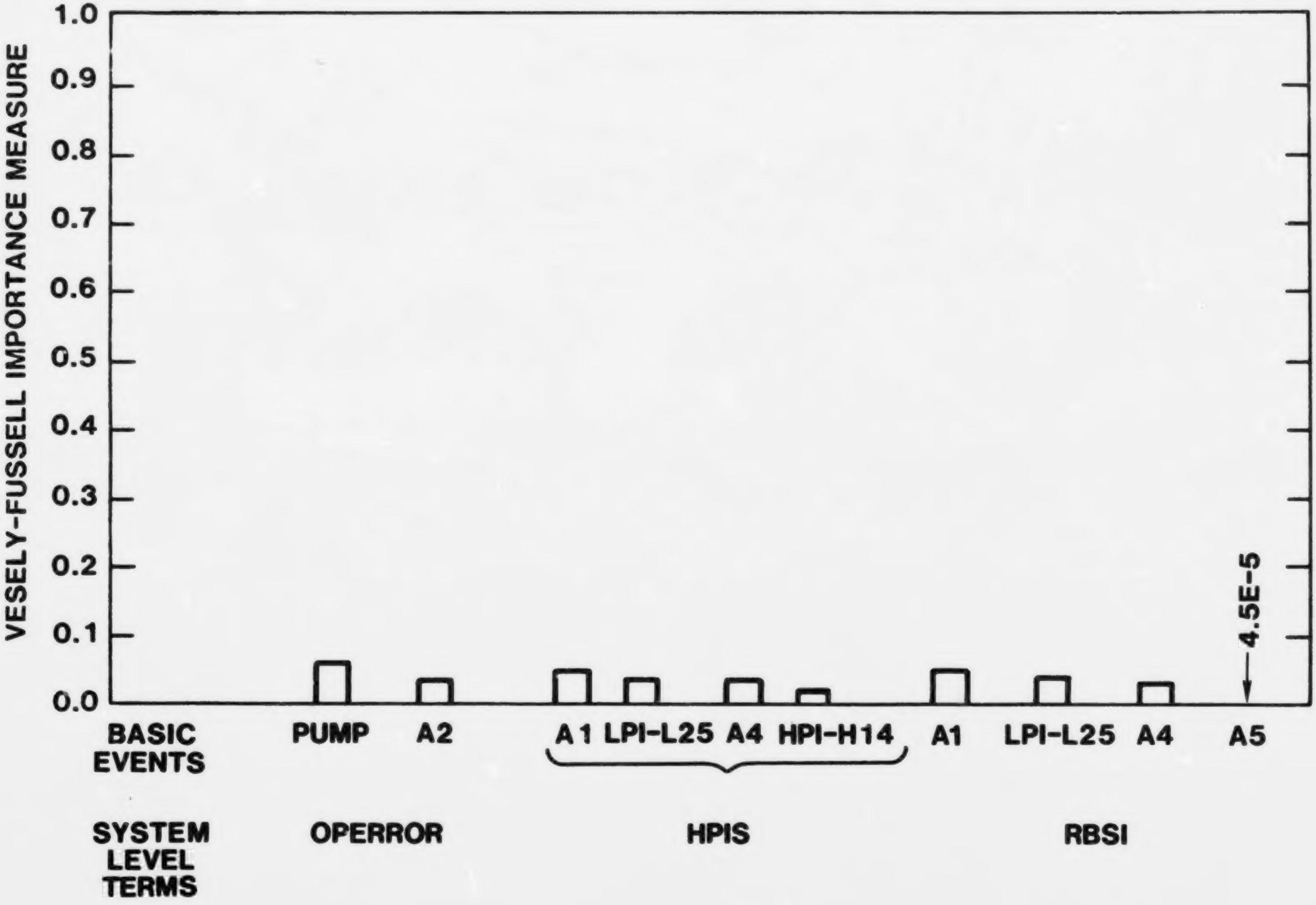

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

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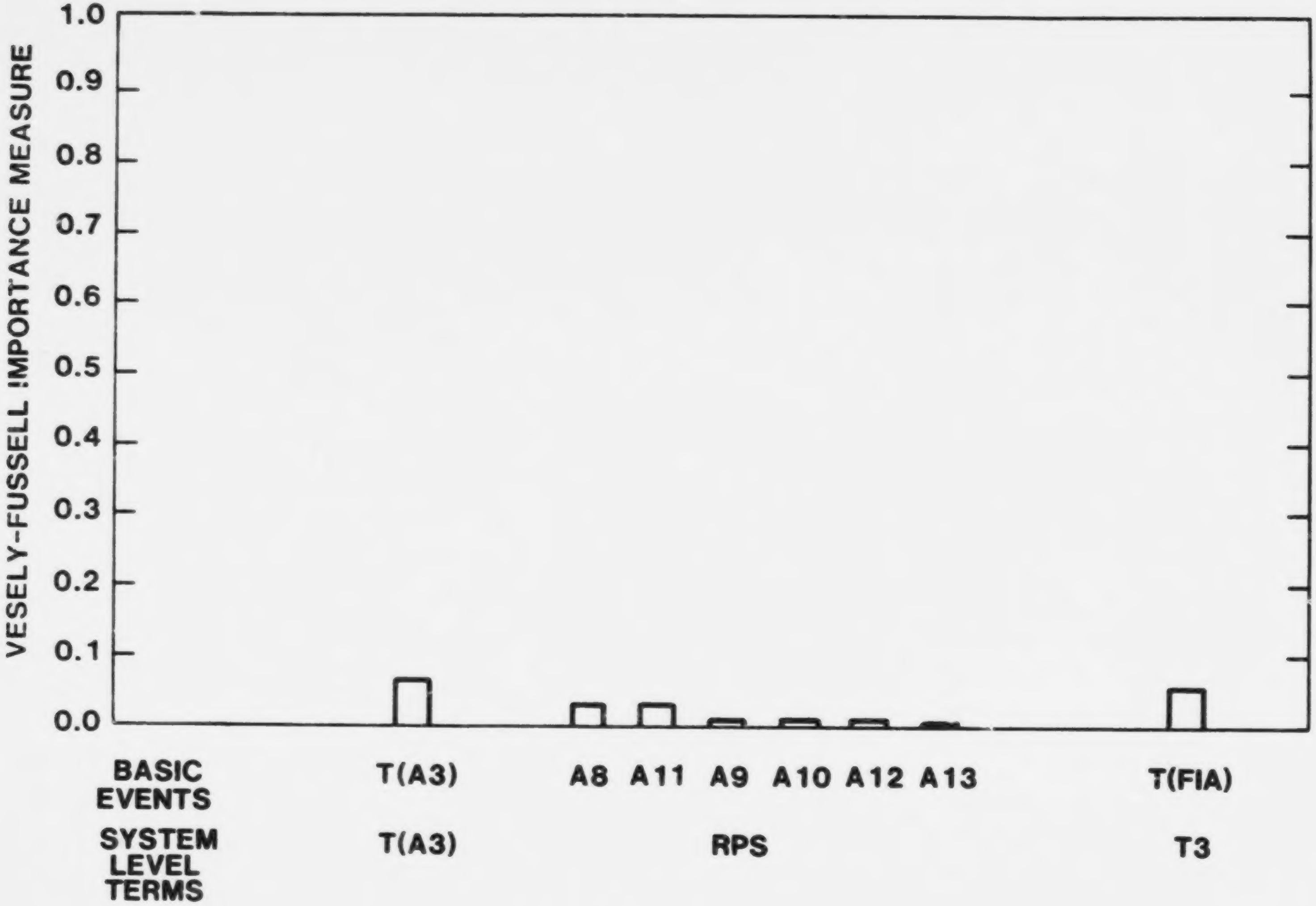

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

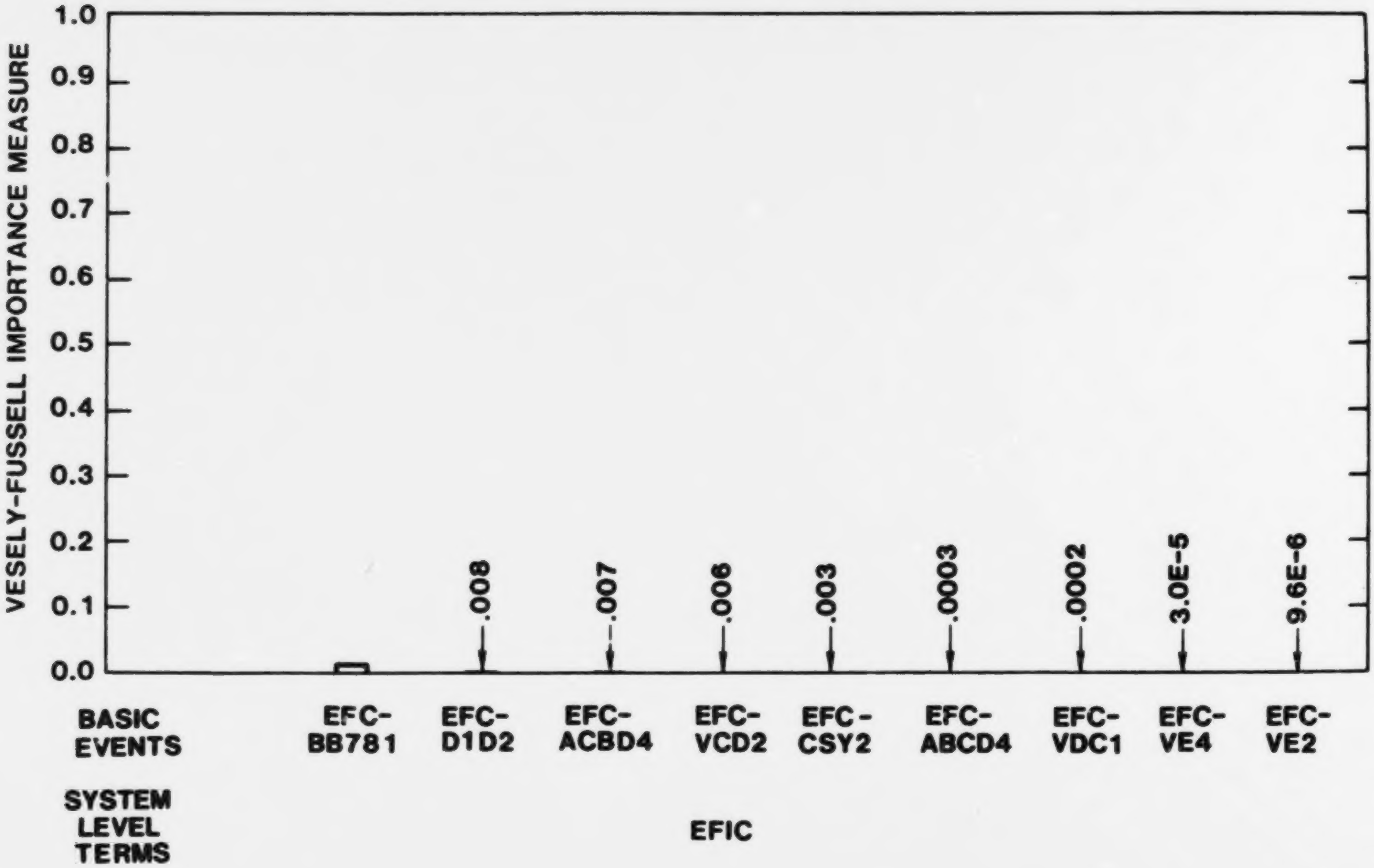

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

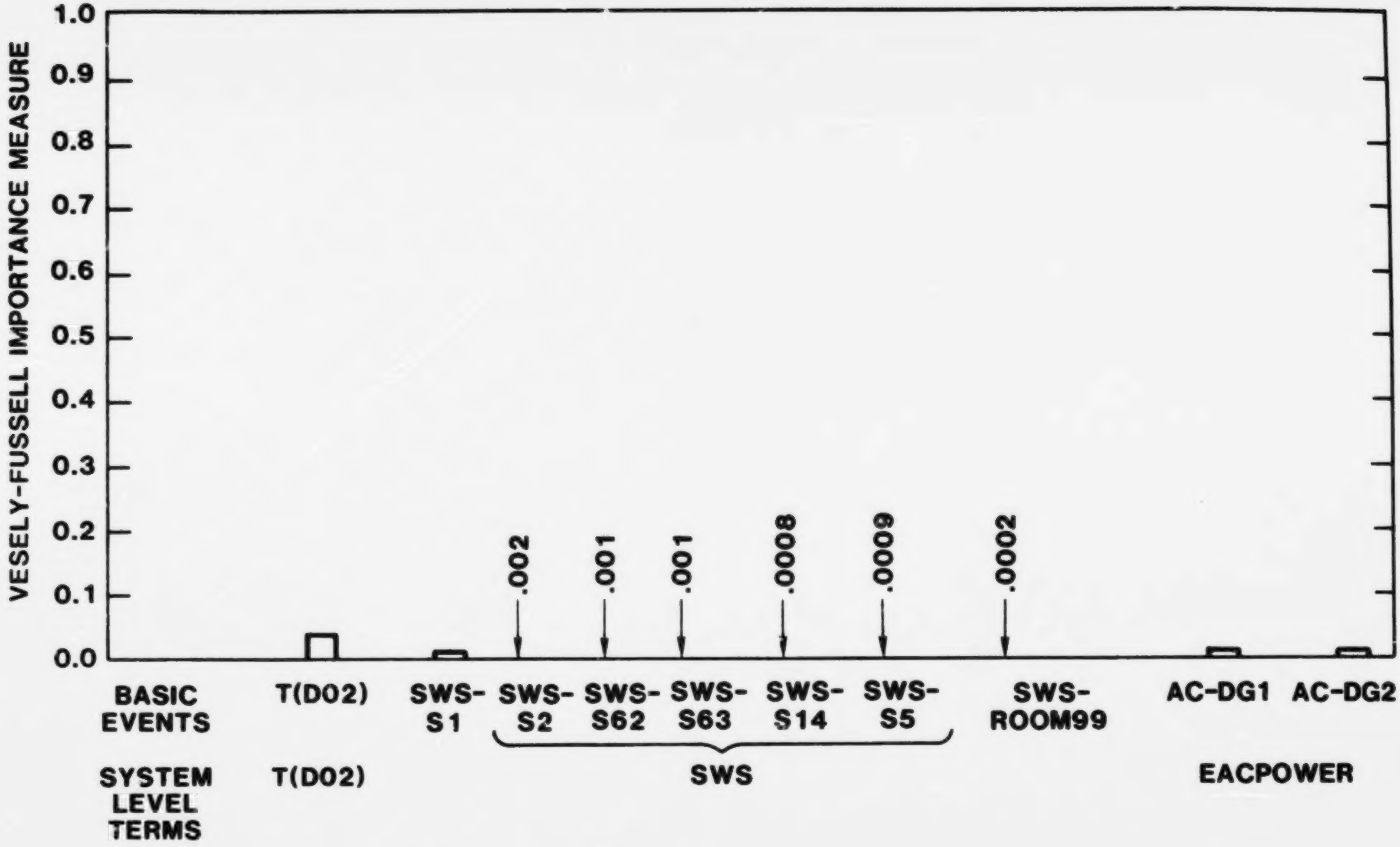

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

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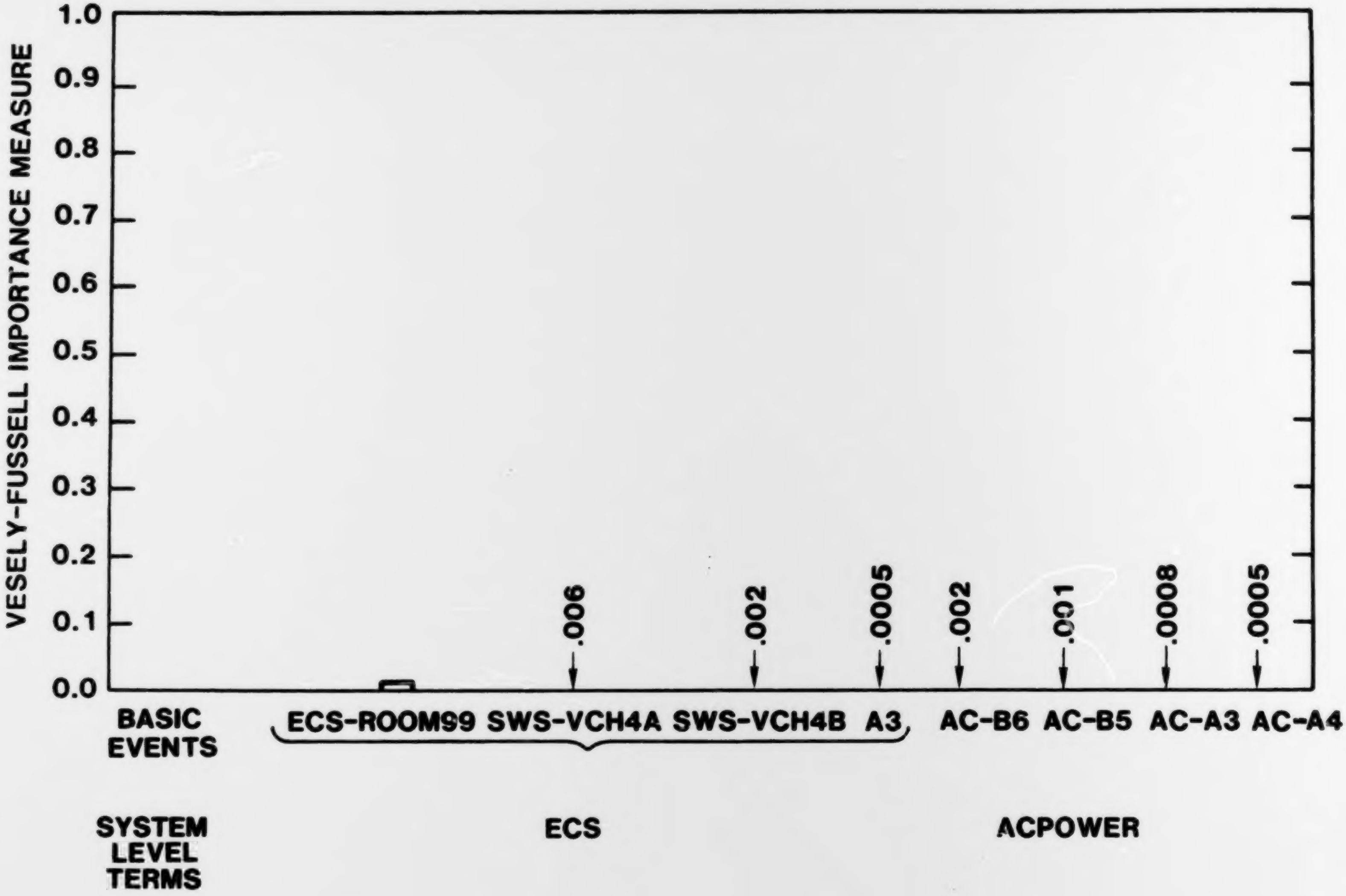

Figure 27. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

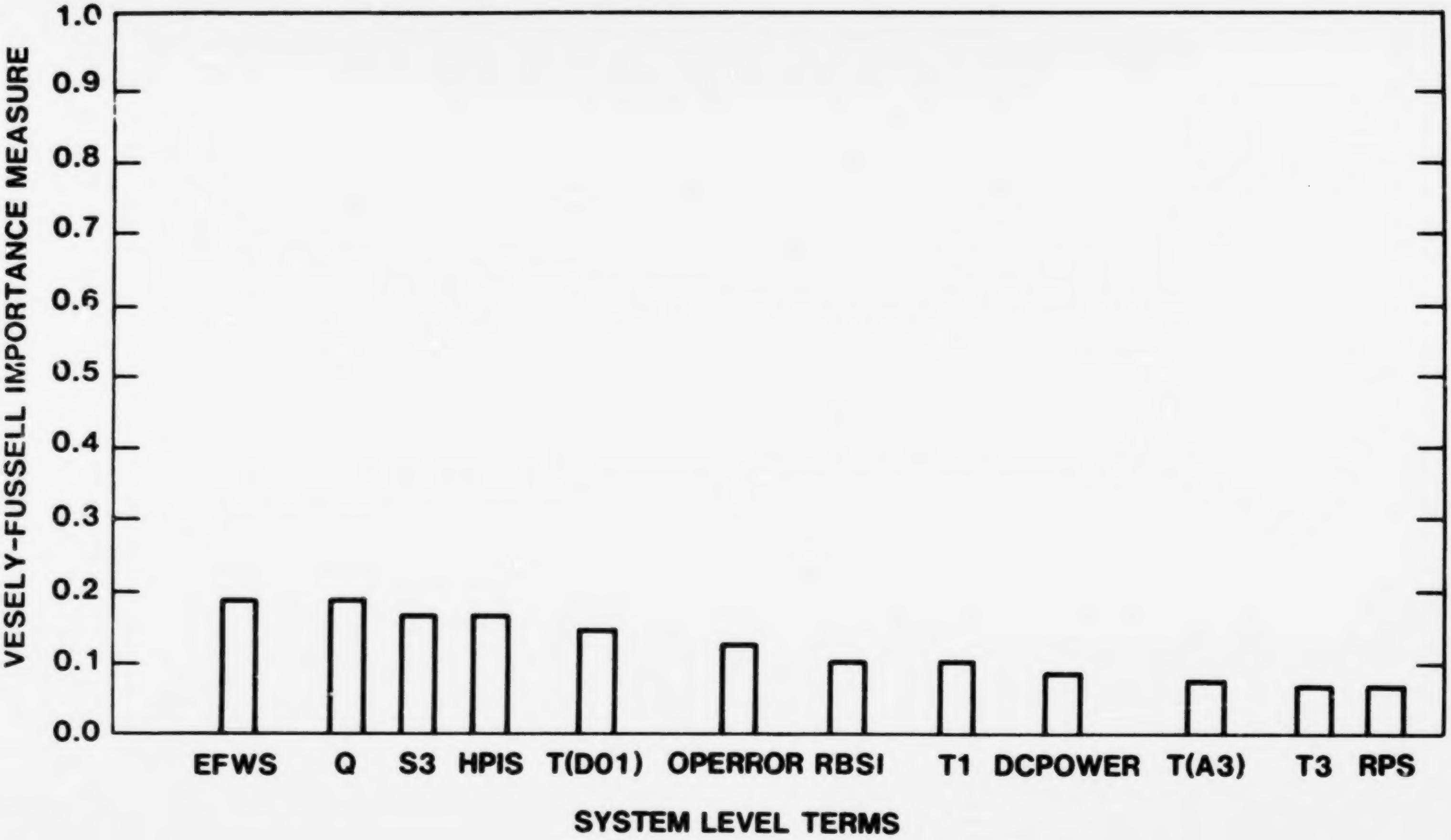

Figure 28. ANO-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

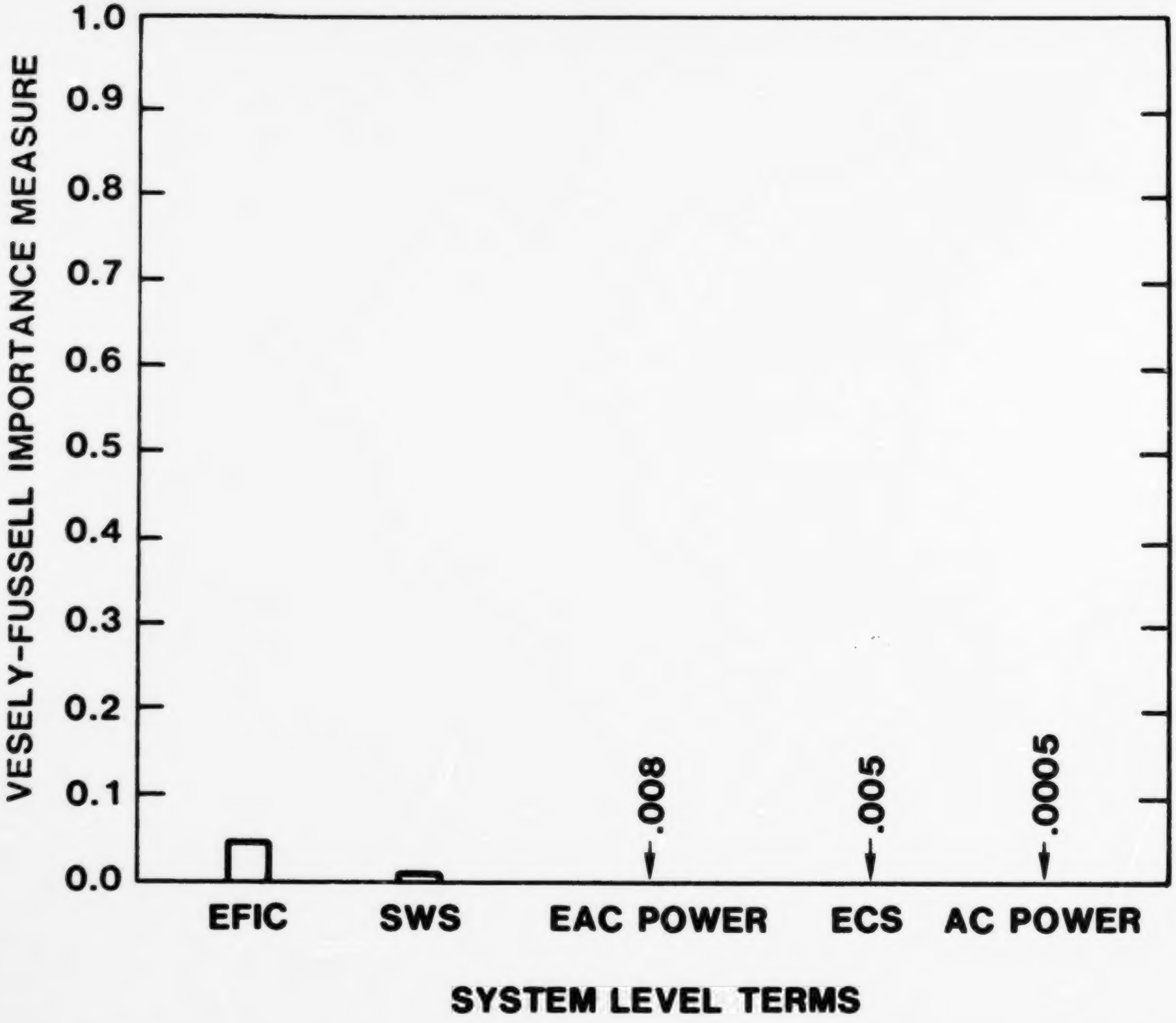

Figure 28. ANO-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Cont)

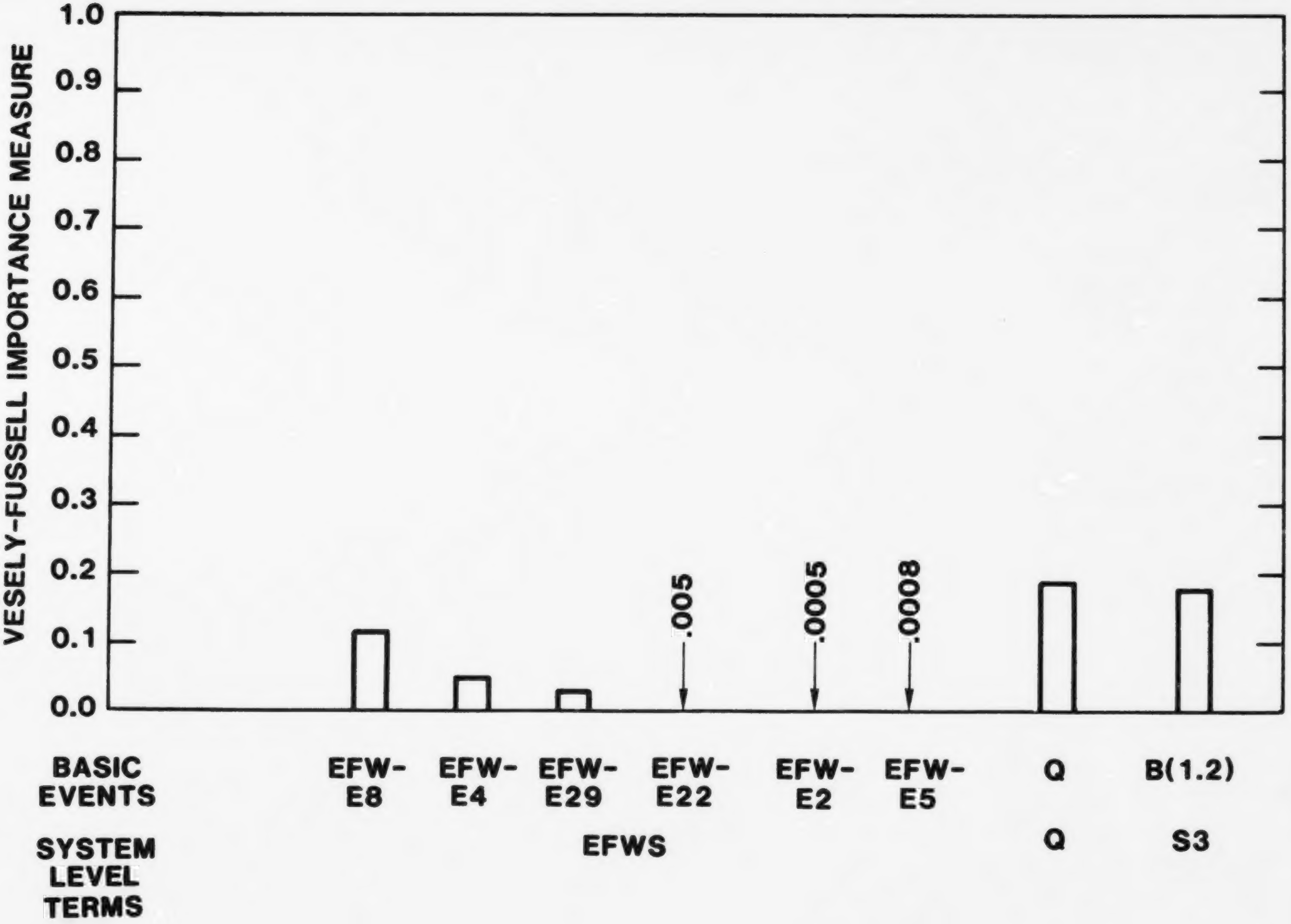

Figure 29. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

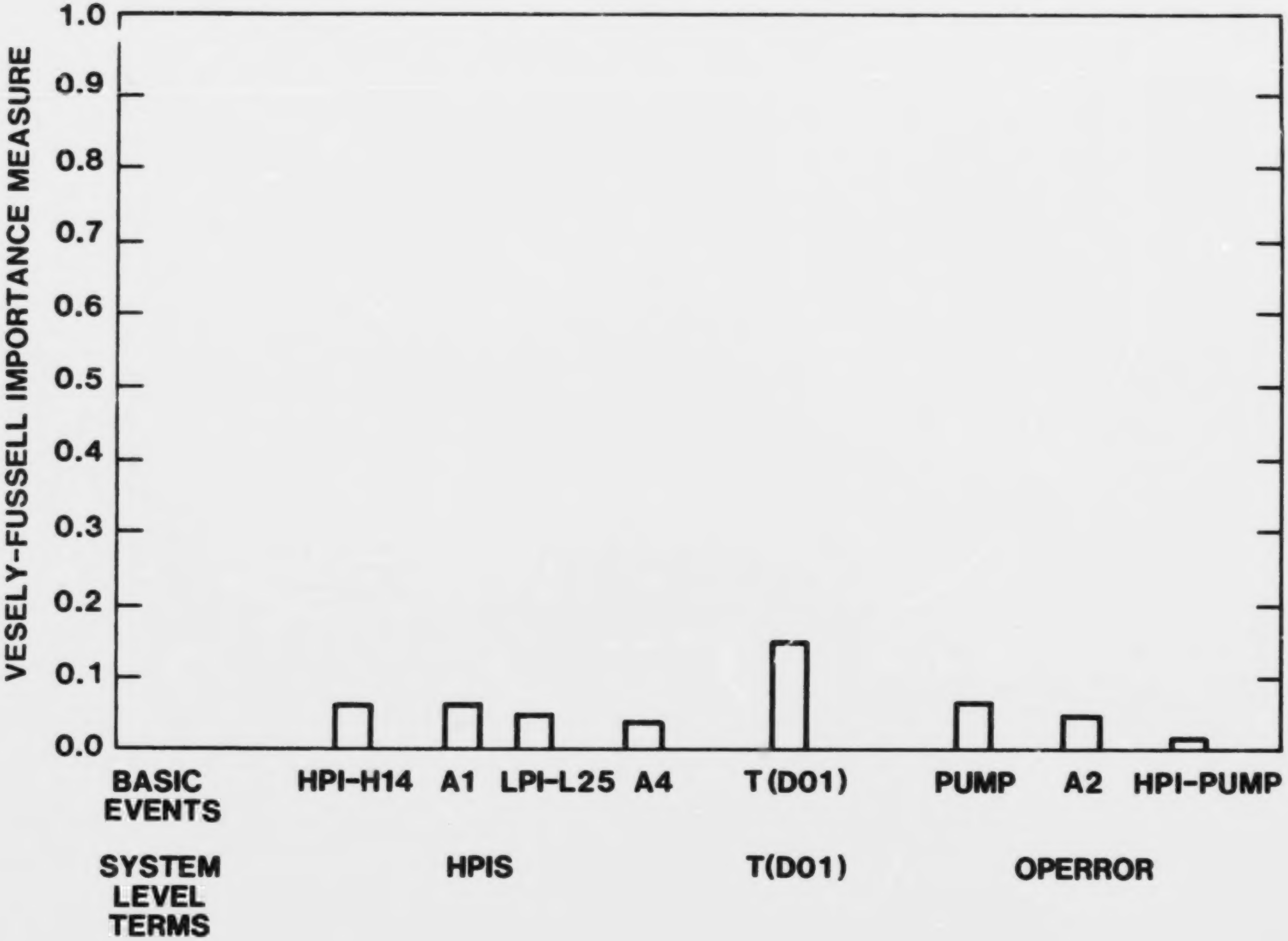

Figure 29. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

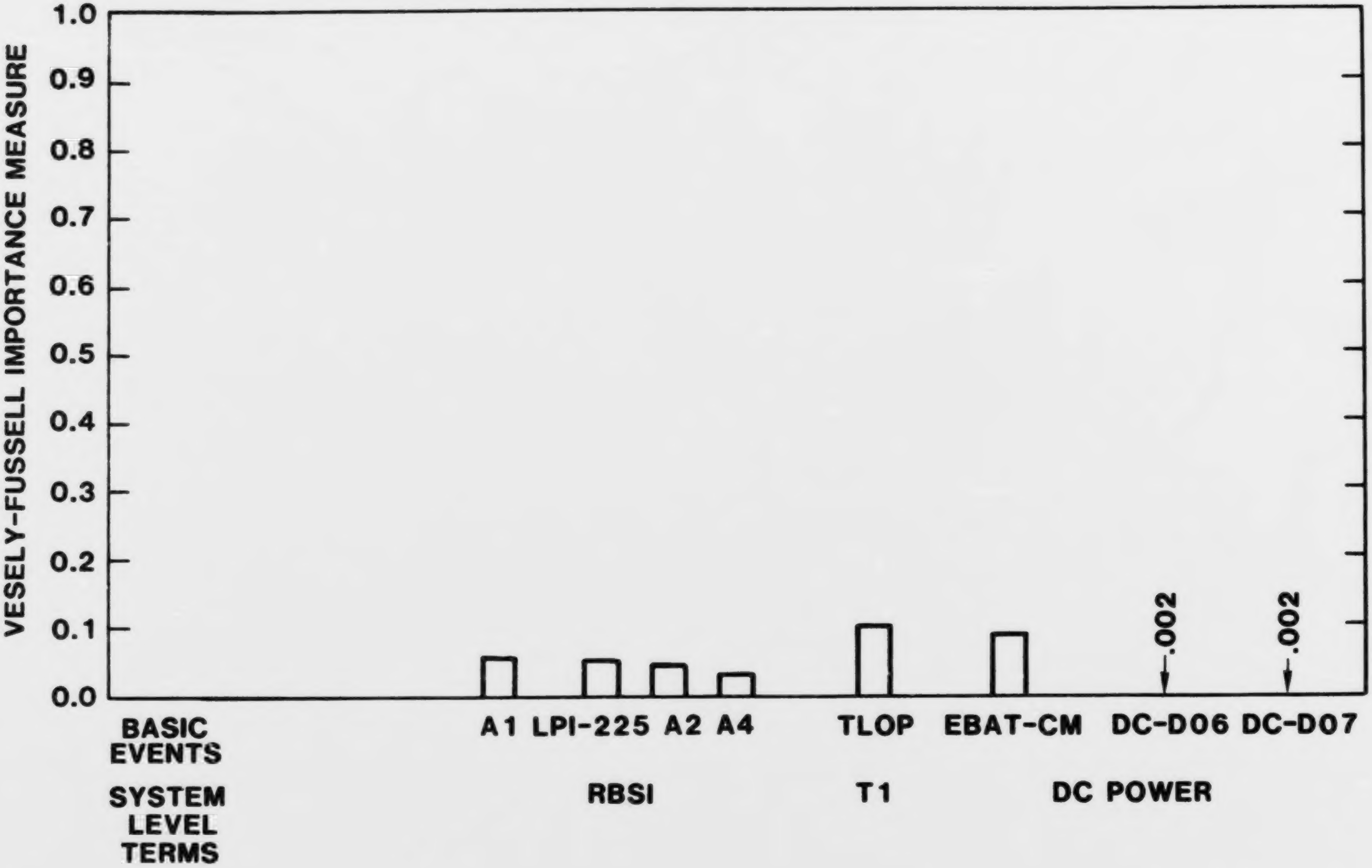

Figure 29. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

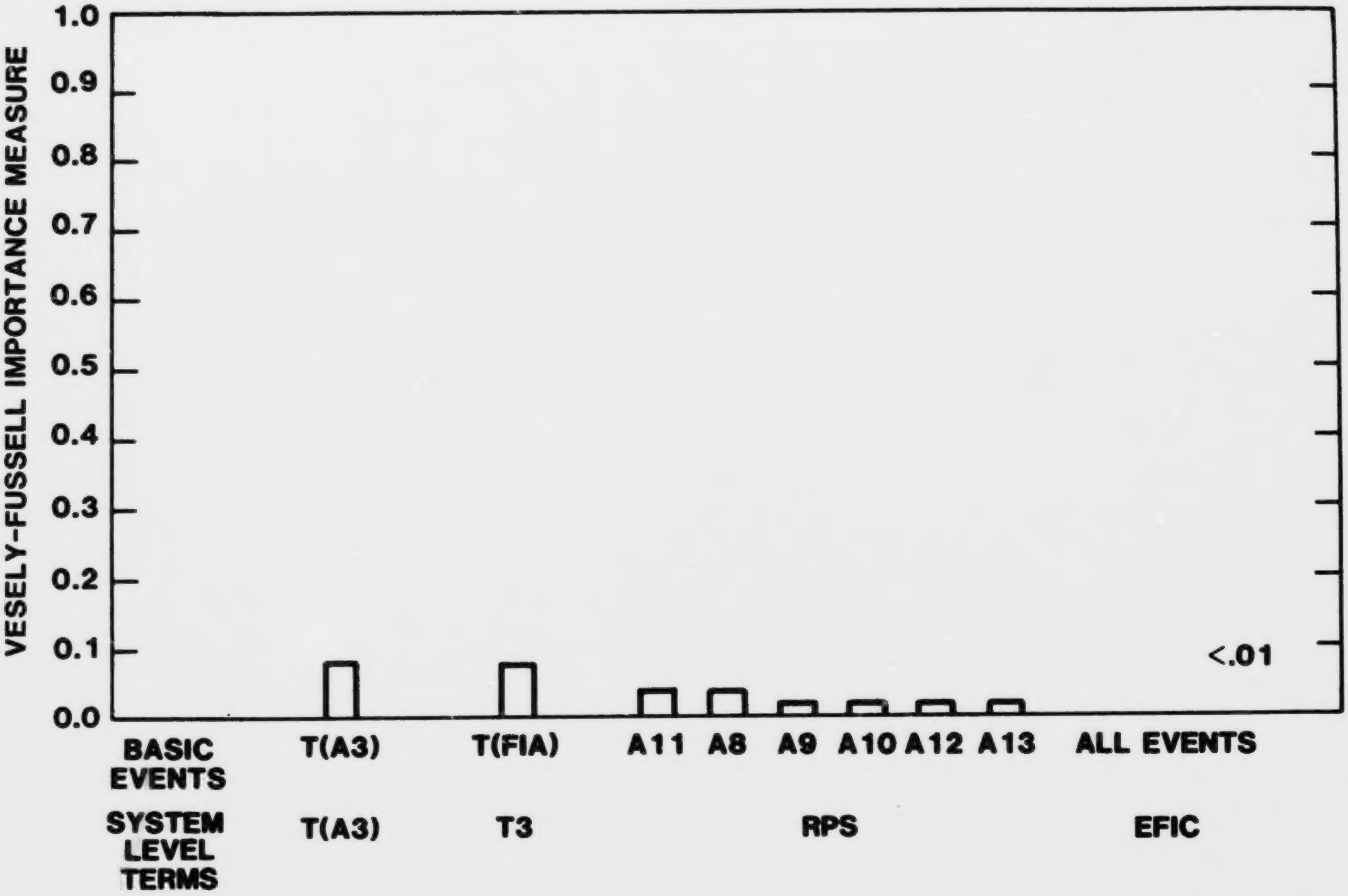

Figure 29. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

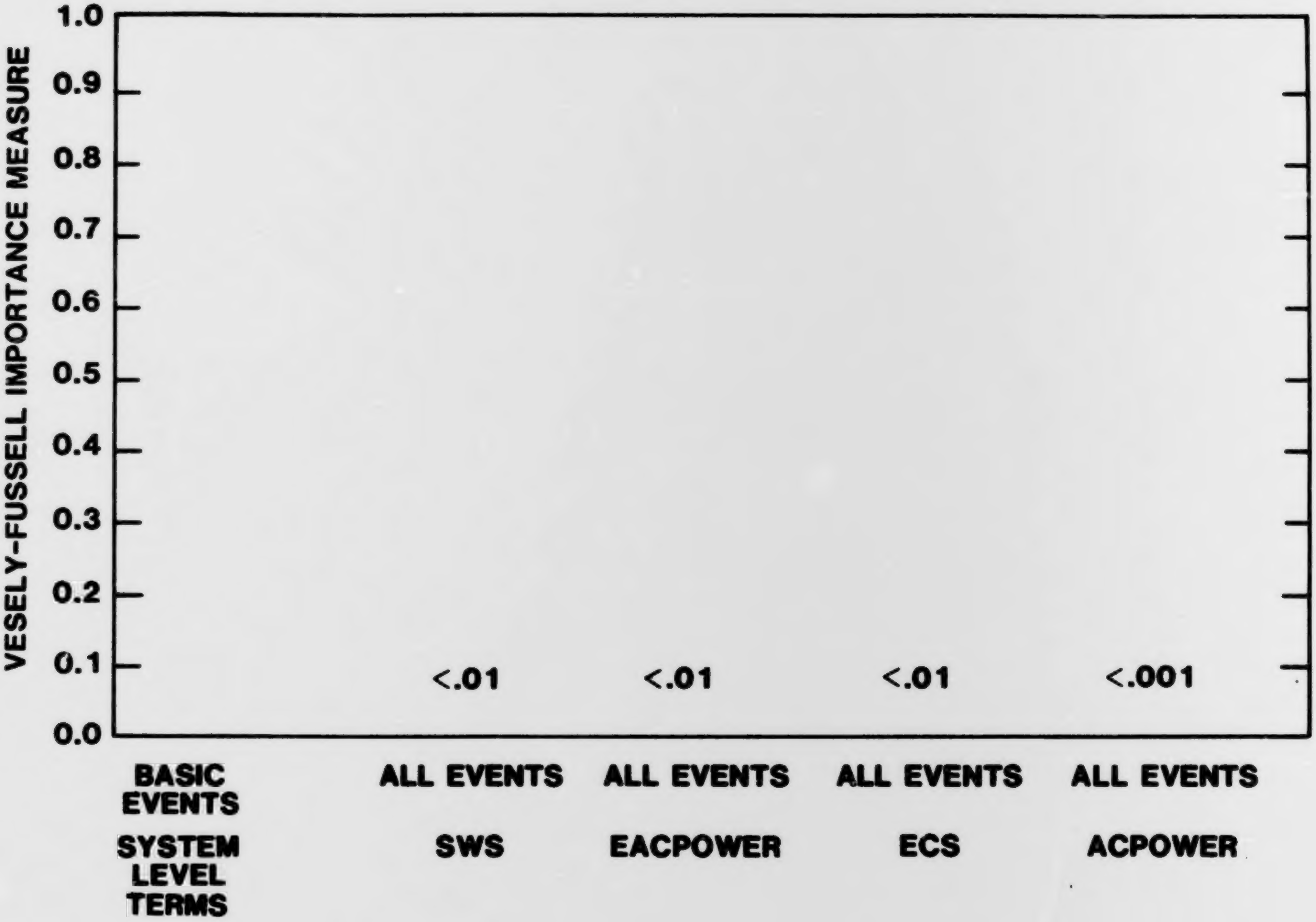

Figure 29. ANO-1, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

3. Criterion 3, Total Plant Risk

The results for Criterion 3 are fairly similar to those for Criterion 2, but some differences do exist. Sequences $T(D01)LD_1YC$ and $T(D02)LD_1YC$ are not dominant to risk and $T(D01)LD_1$ becomes dominant. The slightly smaller release probabilities for $T(D02)LD_1YC$ and $T(D01)LD_1YC$ (Table 60) in category 2 were sufficient to cause these sequences to be nondominant to risk. DC power is no longer the most dominant system. This is due to the relatively low release probability of sequence $T(LOP)LD_1YC$ in category 2. DC power's most dominant term (EBAT-CM) appears only in this sequence. However, while the relative ranking of the basic events has changed somewhat compared to Criterion 2, the same basic events which dominate core melt frequency also dominate plant risk. The results are listed on Tables 60 and 61, and shown on Figures 28 and 29.

c. Oconee-3

The Oconee-3 analysis was done using the Reactor Safety Study Methodology Applications Program (RSSMAP) Report on Oconee-3[15]. This was one of the four RSSMAP studies done by Sandia National Laboratories for the NRC. The cut set information and basic event terms were taken directly from the discussions of the dominant sequences in the Oconee PRA. The cut sets analyzed for the dominant sequences on Table 62 account for 68 percent of the total core melt (Criteria 1 and 2), and 73 percent of the plant risk. About half of the gap between modeled and actual core melt frequency and plant risk is due to limiting dominant sequences to those making up 5 percent or more of total core melt frequency or plant risk. The remaining gap is reflective of the reporting of only dominant cut sets for the sequences in the PRA.

1. Criteria 1 and 2, Total Core Melt Frequency

Eight sequences are dominant to core melt frequency. At the system level (Table 65, Figure 30), operator errors clearly dominate core melt frequency. Event Q and plant system LPSW have about half the importance of operator errors. Plant systems PCS, RPS, LPIS, ESAD, and LPRS have noticeable but smaller contributions to core melt frequency.

At the basic event level (Table 66, Figure 31), Event Q is the single most dominant event followed by failures in the LPSW. Operator error importance is evenly dominated by four different errors. Event Q is also a transient event tree event. Event Q, failure of pressurizer SRVs to reseat, appears in every cut set (Appendix A8) of the two dominant sequences it appears in, T_2MQ-FH and T_2 MQ-H.

The LPSW faults are pumps or valve failures in train A (G1) and train B (F1). These events are defined in Appendix B14 of the PRA. Among the operator errors, HPRSCM and WXCM involve improper operator action to initiate recirculation. LPISCM involves test and maintenance errors which fail LPIS. HPMANK is failure of operator to initiate HPIS during an ATWS sequence. This event is called HPMAN in the PRA ATWS sequences, but is called HPMANK here to differentiate from HPMAN in non-ATWS sequences. The high value of HPMANK reflects the high stress environment during ATWS sequences. These human errors are described in the appendices of the PRA on the systems which they cause to fail.

The Power Conversion System (PCS) and Reactor Protection System (RPS) both have one event which are of equal importance. The RPS event, K, is also the event tree event K, failure of the RPS. The PCS fault, PCSNR, is failure to recover the PCS after a T_2 transient. Recovery of the PCS after T_1 transients (Loss of offsite power) require too many actions to be considered feasible in the short term. This event is described in Section 3.2 of the PRA.

The LPIS and LPRS share significant faults. These are failure of the low pressure pump discharge valves in train A (D) and train B (E). These faults are described in detail in the PRA Appendix A6. The availability of events D and E are significantly greater than other LPIS and LPRS basic events. Sequences S_3H and S_1D appear to be the cause for events D and and E dominating LPIS and LPRS failures. As can be seen by the cut sets of these sequences (Appendix A8), these events appear in two event cut sets in S_1D, and appear together in three event cut sets in S_3H. This, combined with their larger unavailabilities than other low pressure cooling faults, accounts for their dominance. It should be noted that even though the LPIS and LPRS share many of the same basic events, their system importances are not equal. This is because their common events have a larger contribution to injection cooling failure than recirculation failure.

The final event of significance is failure of the Engineered Safeguards Actuation Devices System (ESAD), event CH4. Failure of actuation channel 4 fails LPIS train B and LPSW train A. This event is described in Appendix A10 of the PRA.

2. Criterion 3, Total Plant Risk

Eight sequences dominate plant risk (Table 62). Most of these are the same sequences which dominate core melt frequency. The differences are that sequence S_1D does not dominate risk, and sequence V now becomes dominant. The relative ranking of system importances and of basic event importances is very similar to the results for total core melt frequency. However, LPIS faults and S_1 LOCAs do not contribute to plant risk, and event V, the single event interfacing systems LOCA sequence, not significant to Criterion 2, is now significant.

There is significant dependence among basic events for Oconee as illustrated in the cut sets. However, the four operator errors are independent of each other.

Table 62

Oconee

Plant Vendor	Steam Generator Loops	Containment	MWe Rating	PRA Study
Babcock and Wilcox	2	Dry	886	RSSMAP

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures 1	2	3
1	V	-	-	.13
2	T_2MQ-H	.14	.14	.13
3	S_3H	.12	.12	.12
4	S_1D	.08	.08	-
5	T_2MQ-FH	.06	.06	.08
6	S_3FH	.05	.05	.07
7	T_2MLUO	.10	.10	.09
8	T_2KMU	.10	.10	.09
9	T_1MLUO	.06	.06	.06

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study PWR Release Categories							Sequence Frequency $(R\text{-}yr)^{-1}$
		1	2	3	4	5	6	7	
1	T_2MQ-H	.0001		0.5		.007		0.5	1.1E-5
2	S_3H	.0001		0.5		.007		0.5	1.0E-5
3	S_1D	.01		0.2		.007		0.8	6.7E-6
4	T_2MQ-FH	.0001	0.5	0.5	.007		0.5		5.0E-6
5	S_3FH	.0001	0.5		.007		0.5		4.2E-6
6	T_2MLUO	.0001		0.5		.007		0.5	8.1E-6
7	T_2KMU	.0001		0.5		.007		0.5	7.8E-6
8	T_1MLUO	.0001		0.5		.007		0.5	5.4E-6
9	V		1.0						4.0E-6

Total Plant Core Melt Frequency 8.2E-5

Table 63

Oconee
Boolean Terms From Sequence Fault Trees

SYSTEM	SYSTEM DESCRIPTION	COMPONENT
CSRS	Containment Spray Recirculation	B C W X
EAPOWER	Emergency AC Power System	B3
EFWS	Emergency Feedwater System	CONST2
ESAD	Engineered Safeguards Actuation Devices System	CH1 CH2 CH3 CH4
HPIS	High Pressure Injection System	A1 B1 C1
LOPNRE	Failure to Recover Early from Loss of Offsite Power	LOPNRE
LPIS	Low Pressure Injection System	B C D E
LPRS	Low Pressure Recirculation System	B C D E W X
LPSW	Low Pressure Service Water System	F1 G1
OPERROR	Operator Error	HHMAN HPMAN HPMANK HPRSCM LPISCM RCSRBCM WXCM

Table 63 (Cont.)

SYSTEM	DESCRIPTION	COMPONENT
PCS	Power Conversion System	PCSNR PCST3
Q	Pressurize Safety/Relief Valve Failure to Reclose	Q
RPS	Reactor Protection System	K
S1	Rupture in RCS Piping $10" \leq D \leq 13.5"$	S1
S2	Rupture in RCS Piping $4" \leq D < 10"$	S2
S3	Rupture in RCS Piping $0 < D < 4"$	S3
T1	Loss of Offsite Power Transients	T1
T2	Loss of PCS Due to Transients	T2
T3	Transients That Require Shutdown of PCS	T3
V	Low Pressure Injection System Check Valves Failure	V

Table 64

Oconee RSSMAP
Boolean Terms from Sequence Fault Trees

Boolean Term	Description	Value
A1	Failure of High Pressure Injection System (HPIS) Discharge Valves in Train A	9.8E-3
B	Failure of HPIS Train B Suction Discharge Valves, or Pump	3.5E-2
(B3)	Failure of Both Emergency AC Hydr-electric Generators	5.0E-4
C	Failure of Low Pressure Pump Train A Suction Valves	3.3E-3
CH1	Failure of Actuation Channel 1, Which Fails HPIS Train A	5.0E-3
CH2	Failure of Actuation Channel 2, Which Fails HPIS Train C	5.0E-3
CH3	Failure of Actuation Channel 3, Which Fails Low Pressure Injection System (LPIS), Train A and Low Pressure Service Water (LPSW)	5.0E-3
CH4	Failure of Actuation Channel 4, Which Fails LPIS Train B and LPSW Train A	2.4E-4
CONST1	Failure of Emergency Feedwater System (EFWS) Due Primarily to Hardware Failure of the Turbine Pump Train and Both of the Electric Pump Trains	2.4E-4
CONST2	Failure of the Emergency Feedwater System Due to Failure of Both Electric Pump Trains	6.5E-4
C1	Failure of HPIS Train A Suction Valves	9.8E-3
D	Failure of Low Pressure Pump Train A Discharge Valve	2.2E-4
E	Failure of Low Pressure Pump Train B Discharge Valve	2.2E-4

Table 64 (Cont.)

F1	Pump Failure in Low Pressure Service Water (LPSW) Pump Train B	1.4E-3
G1	Pump Failure in LPSW Pump Train A	1.4E-2
HHMAN	Operator Failure to Manually Start High Head Auxiliary Service Water System	0.1
HPMAN	Operator Failure to Start High Pressure Injection System (HPIS).	1.5E-2
HPMANK	Failure of Operator To Start High Pressure Injection System After an ATWS. This Term is Called HPMAN in the PRA, But is Differentiated Here to Model the High Stress on an Operator During an Accident Involving Boolean Term K	.1
HPRSCM	Failure of Operator to Align Suction of High Pressure Recirculation System to the Discharge of Low Pressure Recirculation System	3.0E-3
K	Failure of the Reactor Protection System Due Primarily to Test and Maintenance	2.6E-5
LOPNRE	Offsite or Onsite AC Power is Not Restored Within Approximately 40 Minutes. This Power is Required to Operate the High Pressure Injection System	0.2
LPISCM	Failure of Low Pressure Injection System Due to Test Valves Left in Wrong Position	3.0E-3
PCSNR	Failure to Restore Power Conversion System (PCS)	0.1
PCST3	Interruption of the PCS After T3 Transient. This Event is Called M in the PRA, But is Given a Different Name Here to Differentiate From M as Used in T1 and T2 Transient Sequences in the PRA, Where it is Incorporated Into the Initiating Transient Event	0.01

Table 64 (Cont.)

Q	Pressurizer Safety/Relief Valve Demanded Open and Failure of at Least One Valve to Reclose	5.0E-4
RCSRBCM	Common Mode Miscalculation of the Sensor/Bistables Which Actuate the HPIS. The Sensor Groups are the Reactor Cooling System (RCS) Low Pressure and Reactor Building High Pressure	3.2E-5
S1	Rupture in the RCS Piping, 10" D 13.5"	1.0E-4/R-yr
S2	Rupture of Reactor Cooling System Piping, 4" D 10"	4.0E-4/R-yr
S3	Rupture in RCS Piping, 0" D 4"	1.3E-3/R-yr
T1	Loss of Offsite Power	.2/R-yr
T2	Loss of Power Conversion System (PCS) After a Transient	3/R-yr
T3	Transients Requiring Shutdown of PCS Initially After Transient	4/R-yr
V	Failure of the Low Pressure Injection System Check Valves When Normally Closed. Isolation MOV is Opened for Quarterly Tests	4.0E-6
W	Failure of Low Pressure Train A Sump Suction Valves	9.6E-3
WXCM	Failure of Operator to Open the Sump Valves at Start of Recirculation Which are Common to Both the Spray and Core Cooling System	3.0E-3
X	Failure of Low Pressure Train A Sump Suction Valves	9.6E-3

Table 65

Oconee

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
OPERROR	.41	.72	.59
T_2	.38	.69	.55
Q	.18	.50	.27
LPSW	.16	.48	.24
S_3	.16	.48	.23
PCS	.10	.42	.14
RPS	.10	.41	.14
S_1	.08	.40	.12
T_1	.07	.38	.10
LPIS	.06	.38	.09
ESAD	.05	.37	.08
LPRS	.04	.35	.05
CSRS	.005	.32	.007

Table 66

Oconee

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
OPERROR	LPISCM	.11	.42	.16
	HPRSCM	.10	.42	.15
	WXCM	.10	.42	.15
	HPMANK	.10	.41	.14
T_2	T_2	.38	.69	.55
Q	Q	.18	.50	.27
LPSW	F1	.16	.48	.24
	G1	.12	.44	.18
S_3	S_3	.16	.48	.23
PCS	PCSNR	.10	.42	.14
RPS	K	.10	.41	.14
S_1	S_1	.08	.40	.12
T_1	T_1	.07	.38	.10
LPIS	D	.05	.37	.08
	E	.05	.37	.08
	C	.005	.32	.007
	B	.005	.32	.007
ESAD	CH4	.05	.37	.07
	CH3	.006	.32	.009
LPRS	D	.05	.37	.08
	E	.05	.37	.08
	W	.01	.32	.02
	X	.01	.32	.02
	C	.005	.32	.007
	B	.005	.32	.007
CSRS	W	.01	.32	.02
	X	.01	.32	.02
	C	.005	.32	.007
	B	.005	.32	.007

Table 67

Oconee

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
OPERROR	.42	.68	.57
T_2	.38	.64	.51
Q	.19	.46	.27
S_3	.17	.43	.23
LPSW	.16	.42	.21
V	.13	.39	.17
PCS	.09	.36	.13
RPS	.09	.36	.12
T_1	.06	.33	.08
ESAD	.04	.31	.05
LPRS	.03	.30	.05
CSRS	.006	.27	.009

Table 68

Oconee

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
OPERROR	WXCM	.13	.40	.18
	HPRSCM	.10	.36	.13
	LPISCM	.10	.36	.13
	HPMANK	.09	.36	.12
T_2	T_2	.38	.64	.51
Q	Q	.19	.46	.27
S_3	S_3	.17	.43	.23
LPSW	F1	.16	.42	.21
	G1	.11	.38	.16
V	V	.13	.39	.17
PCS	PCSNR	.09	.36	.13
RPS	K	.09	.36	.12
T_1	T_1	.06	.33	.08
ESAD	CH4	.04	.31	.05
LPRS	D	.02	.29	.03
	E	.02	.29	.03
	W	.01	.28	.01
	X	.01	.28	.01
	B	.001	.27	.002
	C	.001	.27	.002
CSRS	W	.01	.28	.01
	X	.01	.28	.01
	B	.001	.27	.002
	C	.001	.27	.002

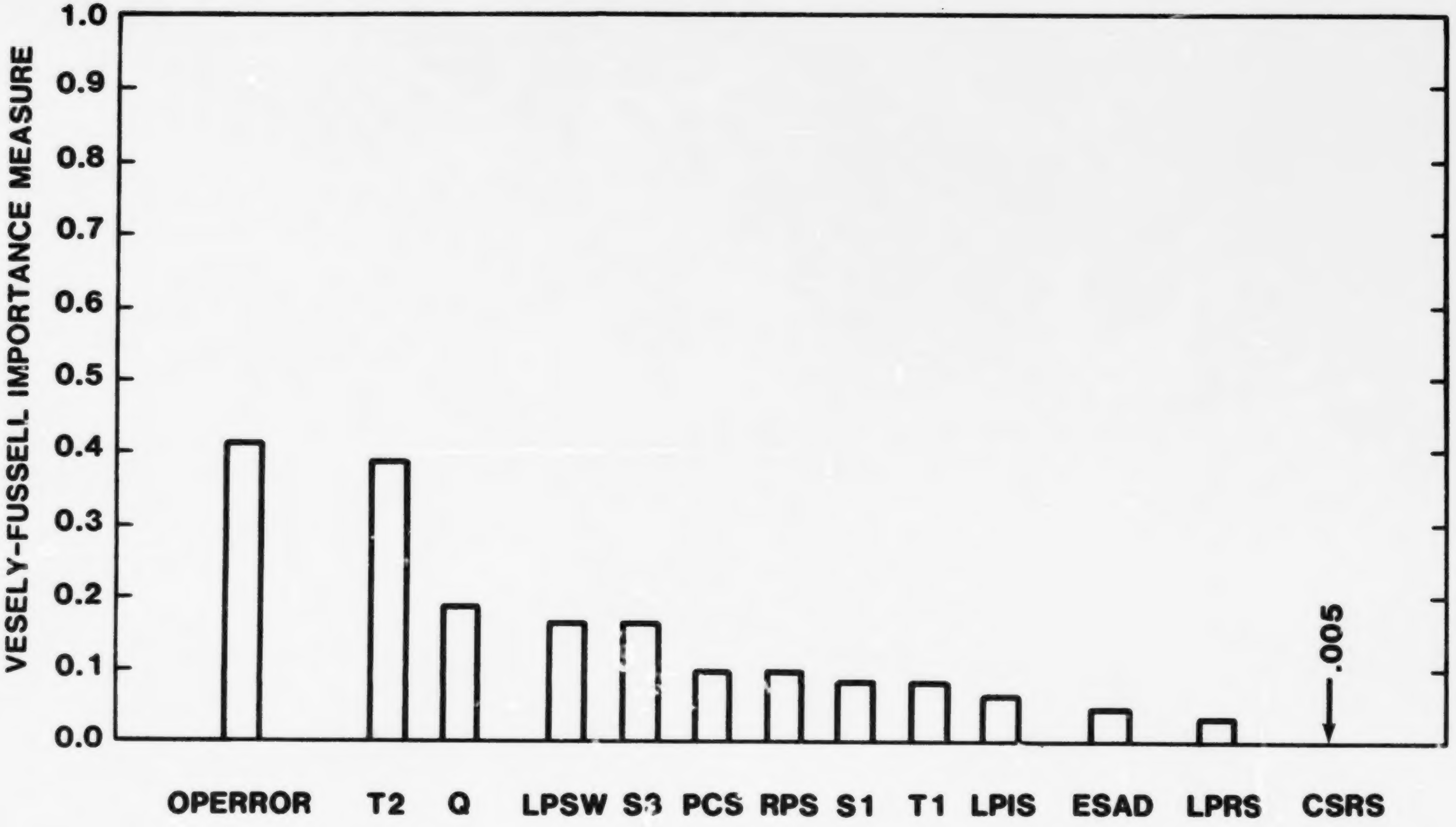

Figure 30. Oconee, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

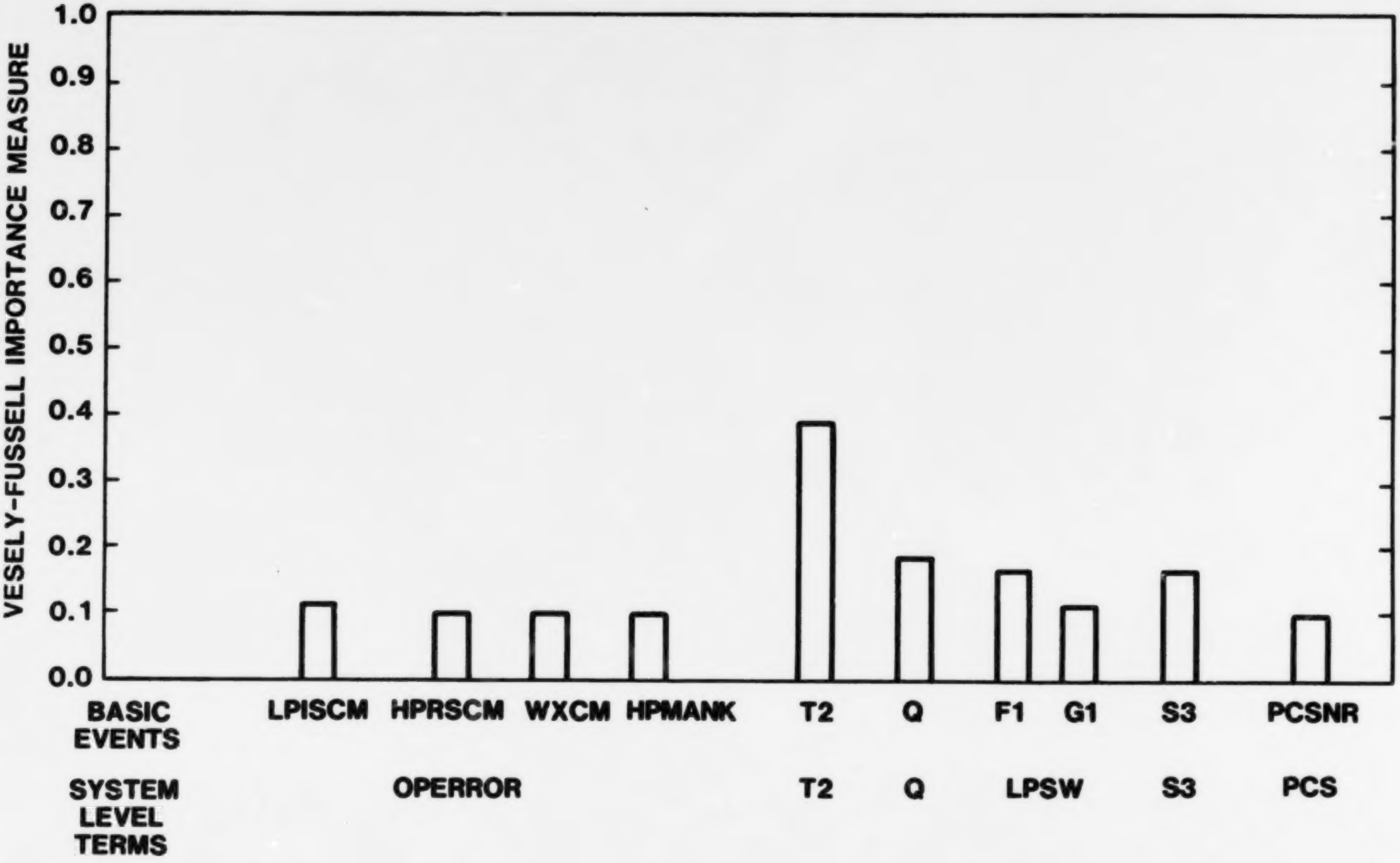

Figure 31. Oconee, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

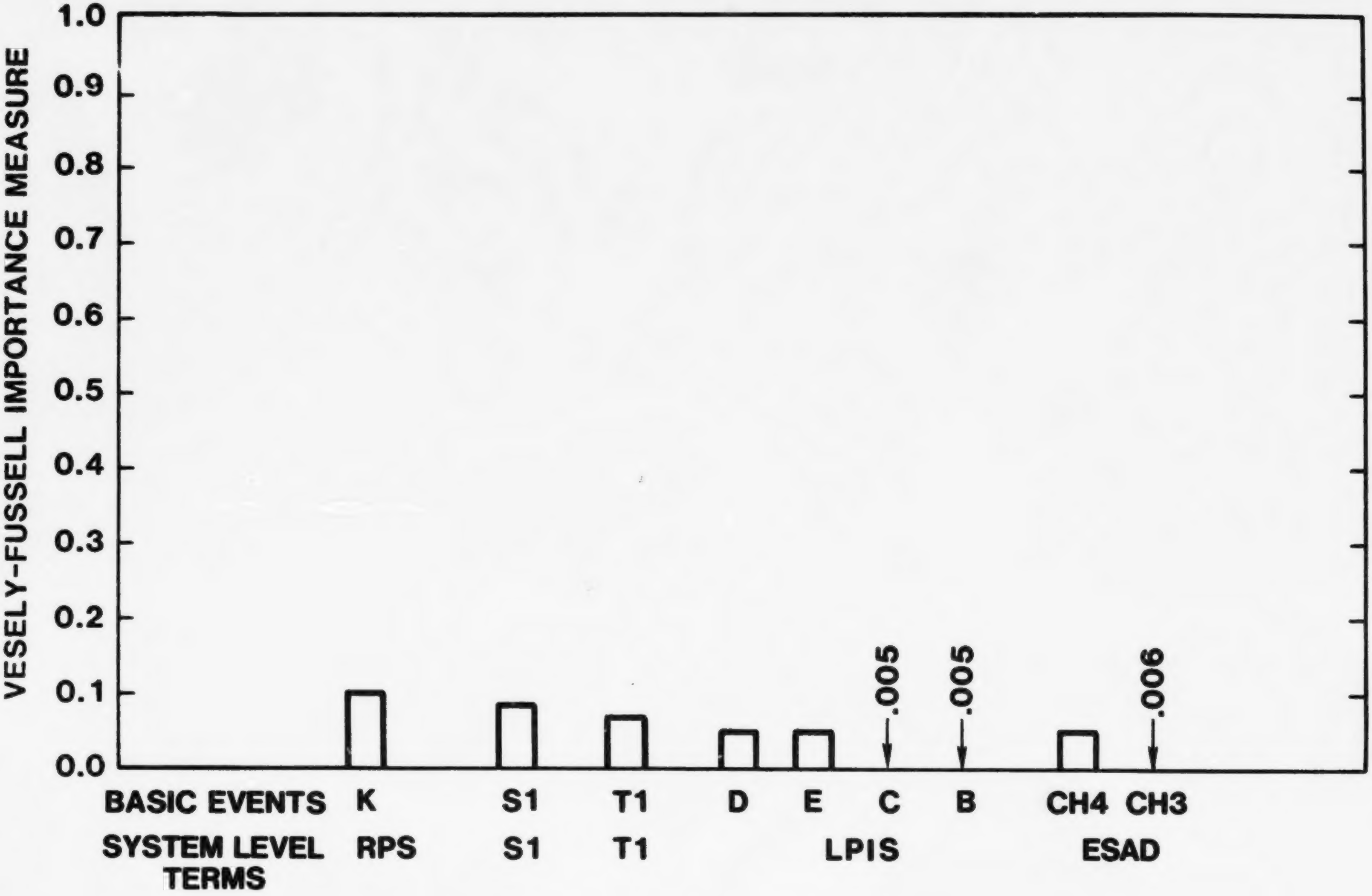

Figure 31. Oconee, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

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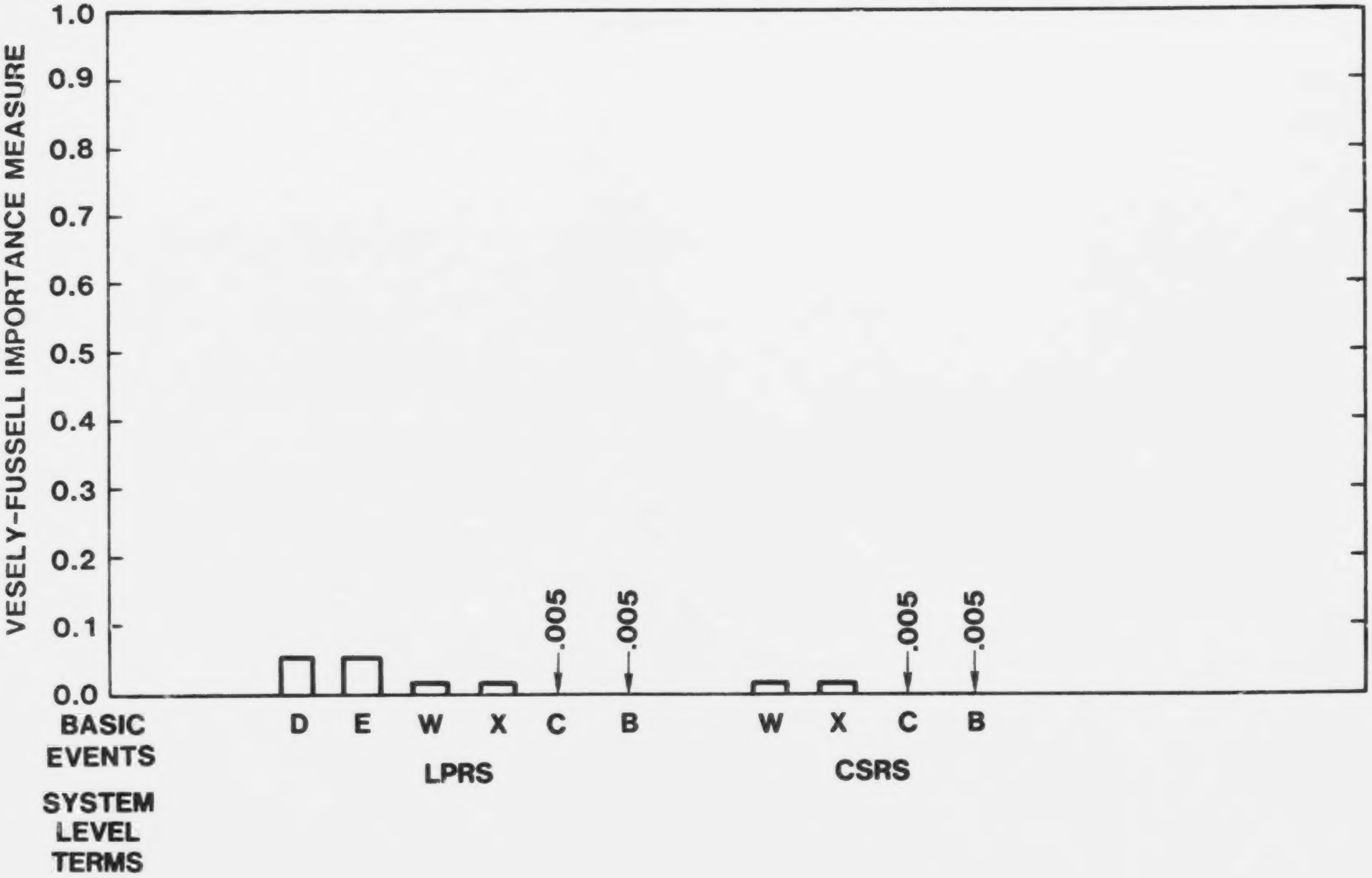

Figure 31. Oconee, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

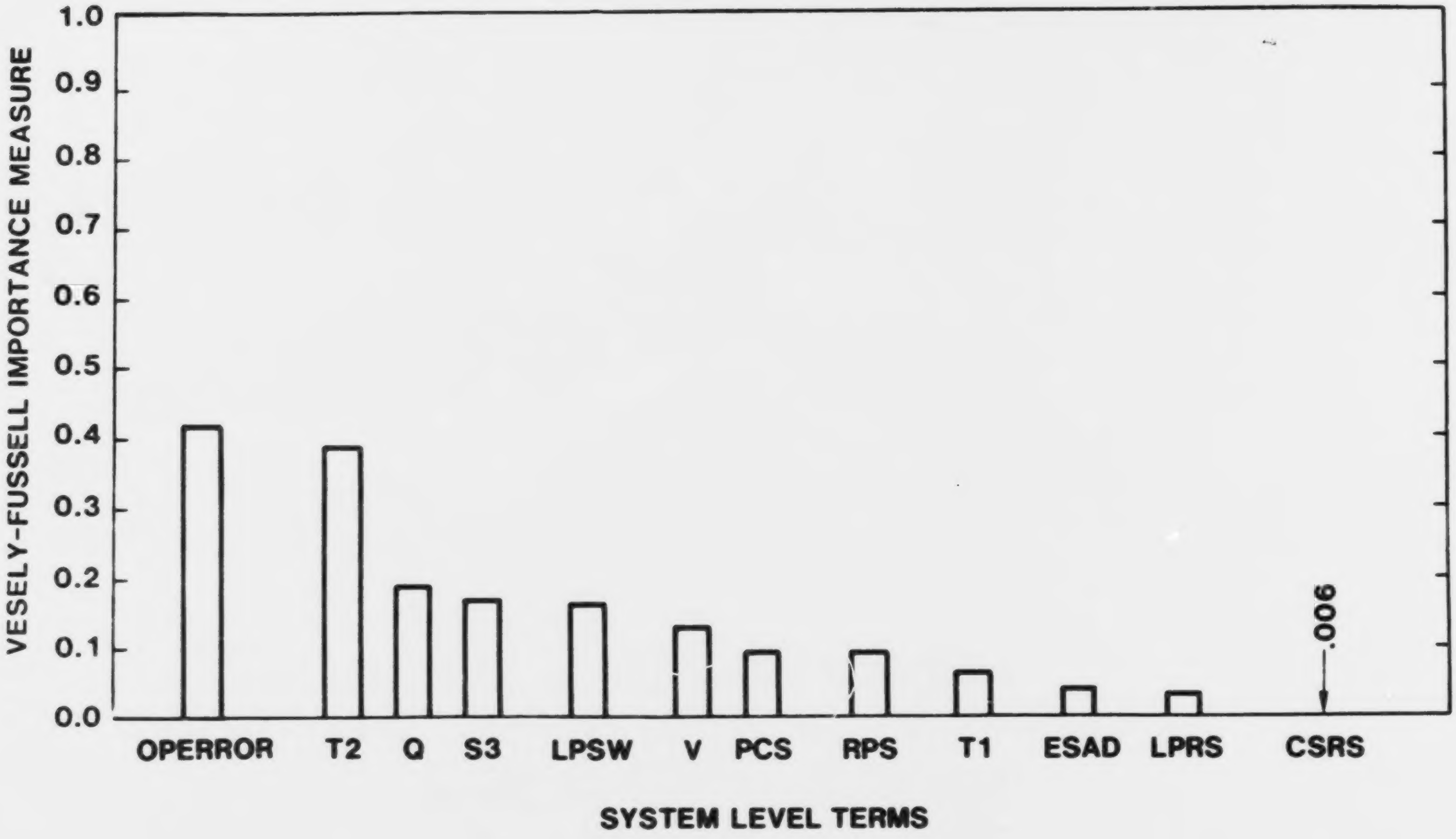

Figure 32. Oconee, Importance of Front Line System and Support Systems to Total Plant Risk: Criterion 3

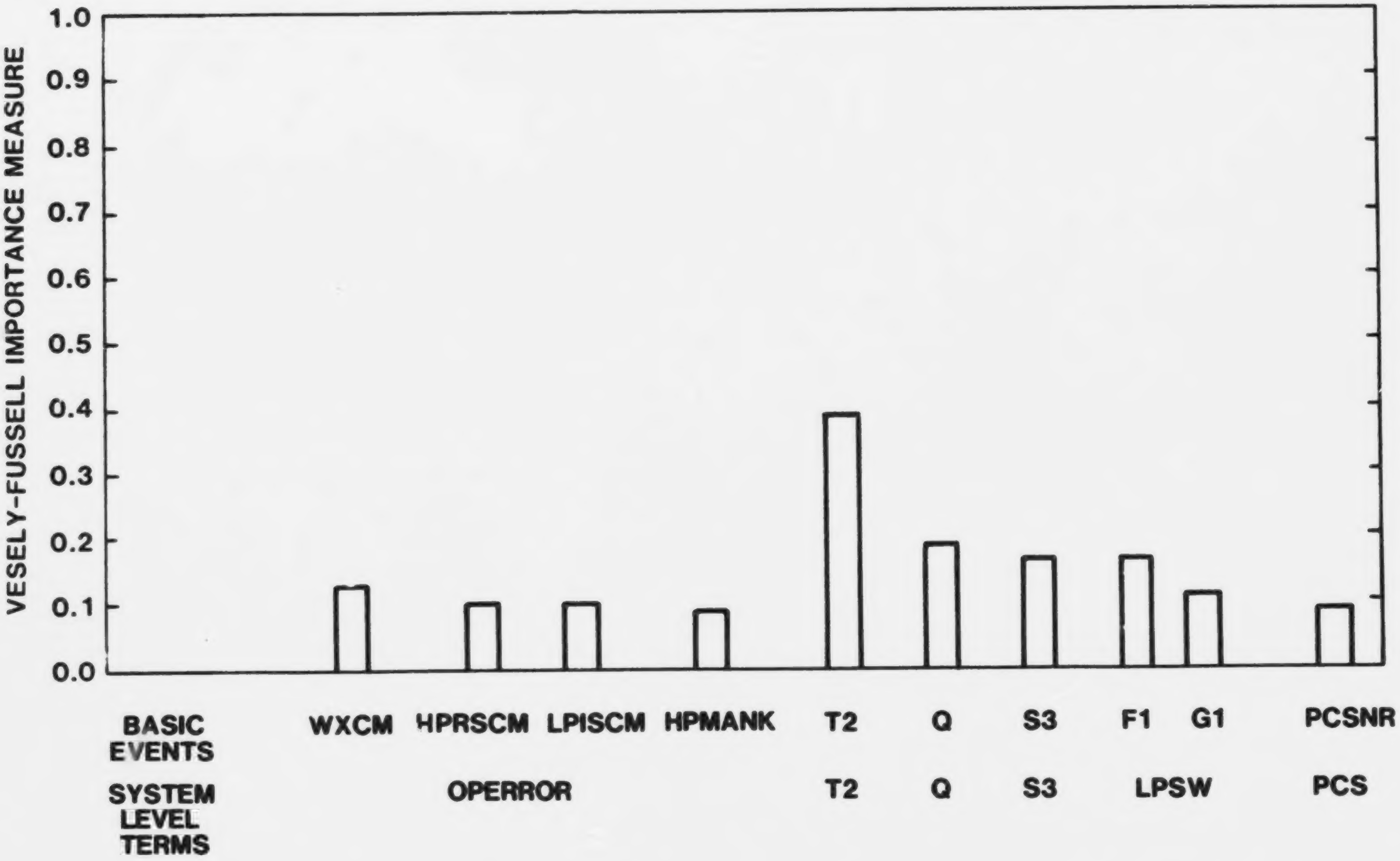

Figure 33. Oconee, Importance of Basic Events Grouped by Systems, to Total Plant Risk: Criterion 3

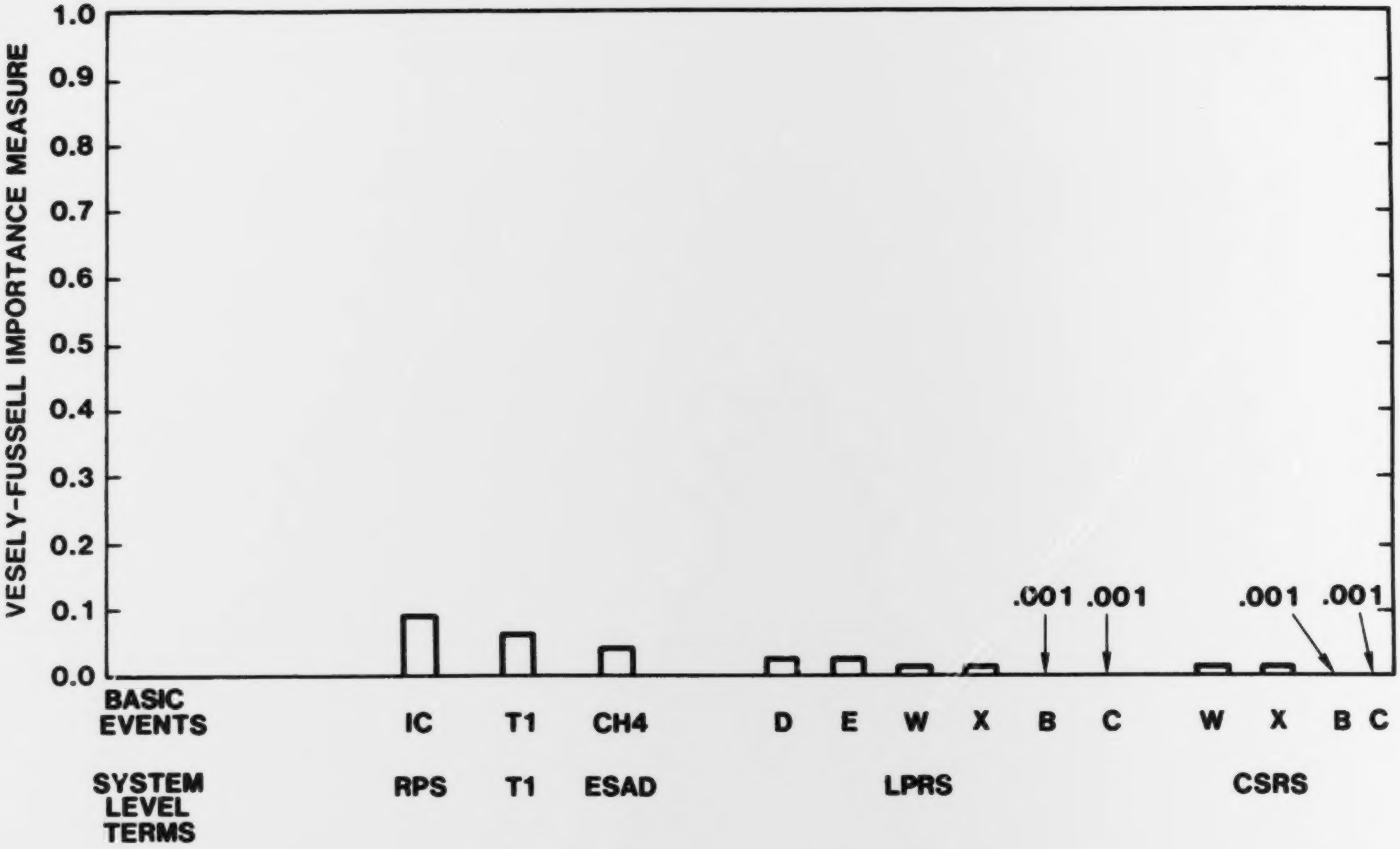

Figure 33. Oconee, Importance of Basic Events Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

3. Combustion Engineering Plant

a. Calvert Cliffs

The analysis for Calvert Cliffs was done using the Reactor Safety Study Methodology Applications Program (RSSMAP) study for Calvert Cliffs.[16] This study was performed for the NRC by Sandia Laboratories. Cut sets and basic events were published for the dominant sequences in the PRA, and are reproduced here in Appendix A9. The cut sets analyzed for the dominant sequences model 83 percent of total core melt frequency (Criteria 1 and 2) and 88 percent of plant risk (Criterion 3).

1. Criteria 1 and 2; Total Core Melt Frequency

Three sequences dominate the core melt frequency. All three sequences are transient sequences, as all the sequences discussed in the PRA as being dominant are transient sequences. The system level results (Table 72, Figure 34) show that only three system level components are significant, the Auxiliary Feedwater System (AFWS), operator errors, and the Power Conversion System (PCS). All other systems have a very small contribution to core melt.

Operator errors and the AFWS share the single most dominant basic event, CONST1 (Table 73, Figure 35). CONST1 is a very general term, comprised of many subevents. Many of the subevents are hardware failures, some are operator errors. CONST1 is described in the AFWS Appendix of the PRA (B.13). In many of the subevents of CONST1 the operator errors and hardware faults are combined into one unavailability, so it is not readily apparent from the importance results as to what amount of the importance of CONST1 is due to operator error, and what is due to hardware faults. However, the single most dominant subevent of CONST1 is operator failure to manually initiate the AFWS (labeled AFWSCM in the PRA Appendix A13). This fault accounts for approximately 1/3 to 1/4 of the AFWS unavailability. The remaining portion of CONST1 unavailability is due to failures of check valves, manual valves, control valves, motor operated valves, and the AFWS turbine pump. However, as stated above, human error has been lumped with these unavailabilities to yield a single value for the event CONST1.

The only other significant system failure is the PCS, which has two events. PCSNR is failure to recover the PCS in the event of a transient which initially involves loss of the PCS (T_2). The Calvert Cliffs PRA uses the Oconee PRA definition and value for PCSNR. PCST3 is failure of the PCS to continue operating following a T_3 transient (PCS initially available). This event is also event tree event M. This event is described in Appendix A2 of the PRA, and is based on WASH-1400 data.

Table 69

Calvert Cliffs

Plant Vendor	Steam Generator Loops	Containment	MWe Reading	PRA Study
Combustion Engineering	2	Dry	850	RSSMAP

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures 1	2	3
1	T_2ML	.45	.45	.43
2	T_1ML	.36	.36	.34
3	T_3ML	.06	.06	.06
4	$T_1MLO(O)$	-	-	.09

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study PWR Release Categories							Sequence Frequency
		1	2	3	4	5	6	7	$(R\text{-}yr)^{-1}$
1	T_2ML	.0001		0.7		.007		0.3	9.0E-4
2	T_1ML	.0001		0.7		.007		0.3	7.2E-4
3	T_3ML	.0001		0.7		.007		0.3	1.2E-4
4	$T_1MLO(O)$	.0001	0.8	0.2	.007				1.0E-4

Total Plant Core Melt Frequency 2.0E-3

Table 70

Calvert Cliffs System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
ACC	High Pressure Injection System Common Mode Failure	HPCM
AFWS	Auxiliary Feedwater System	CONST1
DCPOWER	DC Power System	BATCM
EACPOWER	Emergency AC Power System	D12ST D21ST
ESFAS	Engineered Safeguards Features Actuation System	RASA1 RASB1 RASCM R1 SIA2
HPIS	High Pressure Injection System	B KT LT MT
CCWS	Component Cooling Water System	M1 L1T1 N1 R1
CSRS	Containment Spray Recirculation System	W V
LOPNR	Nonrecovery of Loss of Offsite Power	LOPRL LOPNR8HR NREDF NRLDL
OPERROR	Operator Error	CONST1 NRPORV
PCS	Power Conversion System	PCSNR PCST3
Q	Failure of Pressurizer Safety/Relief Valve to Reseat	Q

Table 70 (Cont.)

System	System Description	Component
RECOVERY	Failure of Recovery Actions to Restore Diesel Generator and Auxiliary Feedwater System	DGNR8HR LOACRES
RMCOOL	Room Coolers	R21 R22
RPS	Reactor Protection System	
SALTWS	Saltwater System	E2T1 F2 N1 SW
T_1	Loss of Offsite Power Initiated Transients	T_1
T_2	Transient Initiated by Loss of Feedwater and Power Conversion System	T_2
T_3	Transient Initiating Event With PCS Initially Available	T_3

Table 71

Calvert Cliffs RSSMAP
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
AFWSCM	Operator Failure to Manually Initiate Auxiliary Feedwater System (AFWS)	1.0E-3
B	Failure of Intake Valves in High Pressure Injection System (HPIS) Train #21 from Refueling Water Storage Tank (RWST).	7.00E-3
BATCM	Failure of All Batteries on Demand Due to Miscalibration of Battery Charger Charging Rate	4.0E-3
C	Failure of Intake Valves in HPIS Train #23 From Refueling Water Storage Tank (RWST)	7.0E-3
CONST1	Failure of Auxiliary Feedwater System (AFWS) Due to Hardware Failures or Operator Errors	2.0E-3
DGNR8HR	Failure to Recover Either Diesel Generator Within 8 Hours	0.7
D12ST	Diesel #12 Unavailability Due to Maintenance and Start Failures. Diesel is Needed to Open Half of All Safety Systems After Loss of Offsite Power.	3.6E-2
D21ST	Diesel #21 Unavailability as in D21ST	3.6E-2
E2T1	Failure of Salt Water System Pump 22 to Restart After Loss of Offsite Power. This Pump Provides Jacket Cooling to Diesel #21.	3.5E-3
F2	Failure of Control Valves in Salt Water System Train #22, Which Provides Jacket Cooling to Diesel #21	2.4E-2
G2T1	Failure of Service Water Pump #22 to Restart After Loss of Offsite Power. This Pump Provides Jacket Cooling to Diesel #21.	3.5E-3

Table 71 (Cont.)

Calvert Cliffs RSSMAP
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
HPCM	Commom Mode Failure of HPIS Due to Flow Diversion Through Failed Low Pressure Injection System (LPIS) or Accumulator Check Valve	8.0E-4
K	Failure of Reactor Protection Systems	2.0E-5
KT	Failure of Manual Valves, Check Valves or Pump in HPIS Train #21.	8.5E-3
LOACRES	Operator Failure to Restore AFWS by Opening Manual Bypass Valves in Steam Admission Line.	0.1
LOPNRL	Failure to Recover Offsite Power Within 3 Hours.	0.1
LOPNR8HR	Failure to Recover Offsite Power in 8 Hours	3.0E-2
LT	Failure of Manual Valves, Check Valves or Pump in HPIS Train #23	8.5E-3
L1T1	Failure of Component Cooling Water System train #21 HPIS Pump Seal Cooling Capability	3.7E-3
MT	Failure of a Normally Closed HPIS Discharge MOV in Train #21	1.3E-2
M1	Failure of Component Cooling System Train #22 HPIS Pump Seal Cooling Capability	9.8E-3
N	Failure of Normally Open MOV on Discharge Side of HPIS Train #23	6.9E-3
NREDF	Failure to Restore Offsite Power Within One Hour so That HPIS Can be Used or Pressurized Safety/Relief Valve (PSRV) Block MOV Can be Closed After at Least One Diesel Has Failed	0.2

Table 71 (Cont.)

Calvert Cliffs RSSMAP
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
NRLDF	Failure to Restore Offsite Power Within 3 Hours so That HPRS Can be Utilized or PSVR Block MOV Can be Closed Given at Least One Diesel Has Failed	0.1
NRORV	Failure of the Operator to Close PSRV Block Valve Given That Diesels Have Started	0.1
N1	Same Failure as SW, but in Train #22	2.4E-2
PCSNR	Nonrecovery of Power Conversion System (PCS)	0.1
PCST3	Interruption of PCS Given a T_3 Transient. This Term is Labelled M in the PRA, but is Labelled Differently Here to Avoid Confusion With Event M of the Event Tree.	0.01
Q	Failure of PSRV to Reseat After Being Demanded Due to T_1 Transient	8.0E-2
QT2	Failure of PSRV to Reseat After Being Demanded Due to T_2 Transient	8.0E-2
RASA1	Failure of Actuation Subchannel A1	5.0E-3
RASB1	Failure of Recirculation Actuation Channel Subchannel B1	5.0E-3
RASCM	Failure of All Containment Spray Recirculation System (CSRS) and High Pressure Recirculation System (HPRS) Pumps by Common Mode Failure of Actuation System	1.0E-3
R1	Failure of Control Valves in Component Cooling Water System Needed to Cool HPRS Pump Seals	0.1
R21	Failure of Room Cooler #21, Which Fails Train #21 of CSRS and #22 of HPRS	1.9E-2

Table 71 (Cont.)

Calvert Cliffs RSSMAP
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
R22	Failure of Room Cooler #22, Which Fails CSRS Train #23 and HPRS Train #23	2.5E-2
SIA2	Failure of Safety Injection Actuation Subchannel A2 to Initiate HPIS pump #21	5.0E-3
SW	Failure of Control Valves in Salt Water Systems Train #21 Needed From HPRS Pump Seal Cooling. This Term is Labelled as S_1 in the PRA, but is Relabelled Here to Avoid Confusion With the LOCA Event S_1.	2.4E-2
T_1	Transient Initiated by Loss of Offsite Power.	0.2/R-yr
T_2	Transient Initiated by Loss of Feedwater and Power Conversion System (PCS).	3/R-yr
T_3	Transient Initiating Event With PCS Initially Available	4/R-yr
V	Failure of Sump Recirculation Valves to HPRST Train #23 and CSRS Train #23	1.3E-2
W	Failure of Sump Recirculation Valves to HPRST Train #22 and CSRS Train #21	1.3E-2

Table 72

Calvert Cliffs

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criterion 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
AFWS	.81	.98	.97
OPERROR	.81	.98	.97
PCS	.51	.68	.61
T_2	.45	.62	.54
T_1	.32	.49	.39
T_3	.06	.23	.07
EACPOWER	.02	.19	.03
RECOVERY	.02	.19	.03
SALTWS	.009	.18	.01

Table 73

Calvert Cliffs

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criterion 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Initial Importance	Worst Case Importance	Proportional Importance
AFWS	CONST1	.81	.98	.97
OPERROR	CONST1	.81	.98	.97
PCS	PCSNR	.45	.62	.54
	PCST3	.06	.23	.07
T_2	T_2	.45	.62	.54
T_1	T_1	.32	.49	.39
T_3	T_3	.06	.23	.07
EACPOWER	D12ST	.02	.19	.03
	D21ST	.01	.18	.02
RECOVERY	LOACRES	.02	.19	.03
SALTWS	F2	.009	.18	.01

Table 74

Calvert Cliffs

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Initial Importance	Worst Case Importance	Proportional Importance
AFWS	.77	.90	.88
OPERROR	.77	.90	.88
PCS	.49	.61	.56
T_2	.43	.55	.49
T_1	.39	.51	.44
DCPOWER	.07	.19	.08
T_3	.06	.18	.07
RECOVERY	.03	.16	.37
EACPOWER	.03	.16	.37
SALTWS	.01	.14	.01
LOPNR	.01	.13	.01

Table 75

Calvert Cliffs

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk Frequency: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
AFWS	CONST1	.77	.90	.88
OPERROR	CONST1	.77	.90	.88
PCS	PCSNR	.43	.55	.49
	PCST3	.06	.18	.07
T_2	T_2	.43	.55	.49
T_1	T_1	.39	.51	.44
DCPOWER	BATCM	.07	.19	.08
T_3	T_3	.06	.18	.07
RECOVERY	LOACRES	.02	.15	.03
	DGNR8HR	.008	.13	.009
EACPOWER	D12ST	.03	.15	.03
	D21ST	.02	.15	.03
SALTWS	F2	.01	.14	.01
LOPNR	LOPNR8HR	.008	.13	.009
	LOPNRL	.004	.13	.004

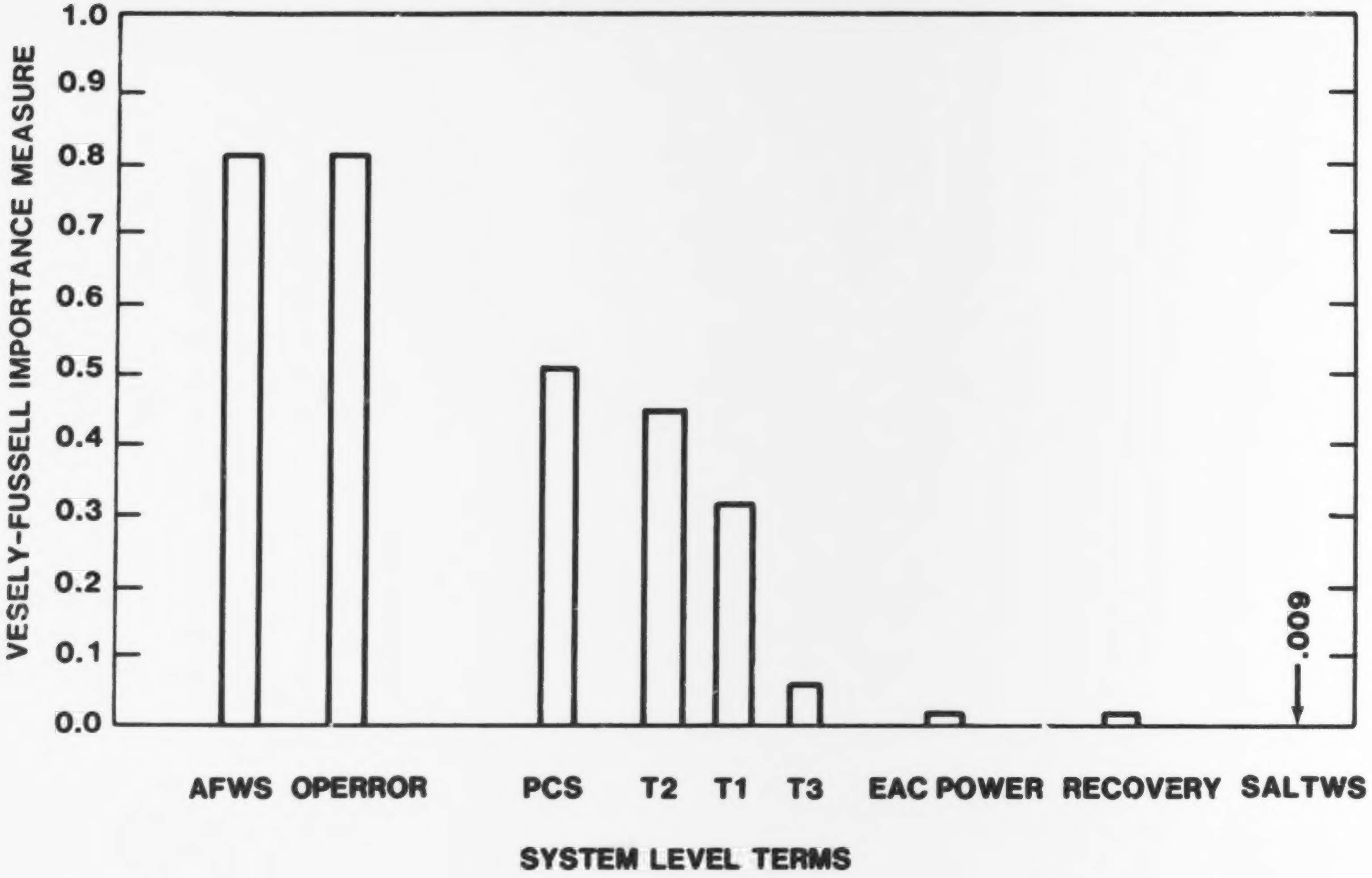

Figure 34. Calvert Cliffs, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

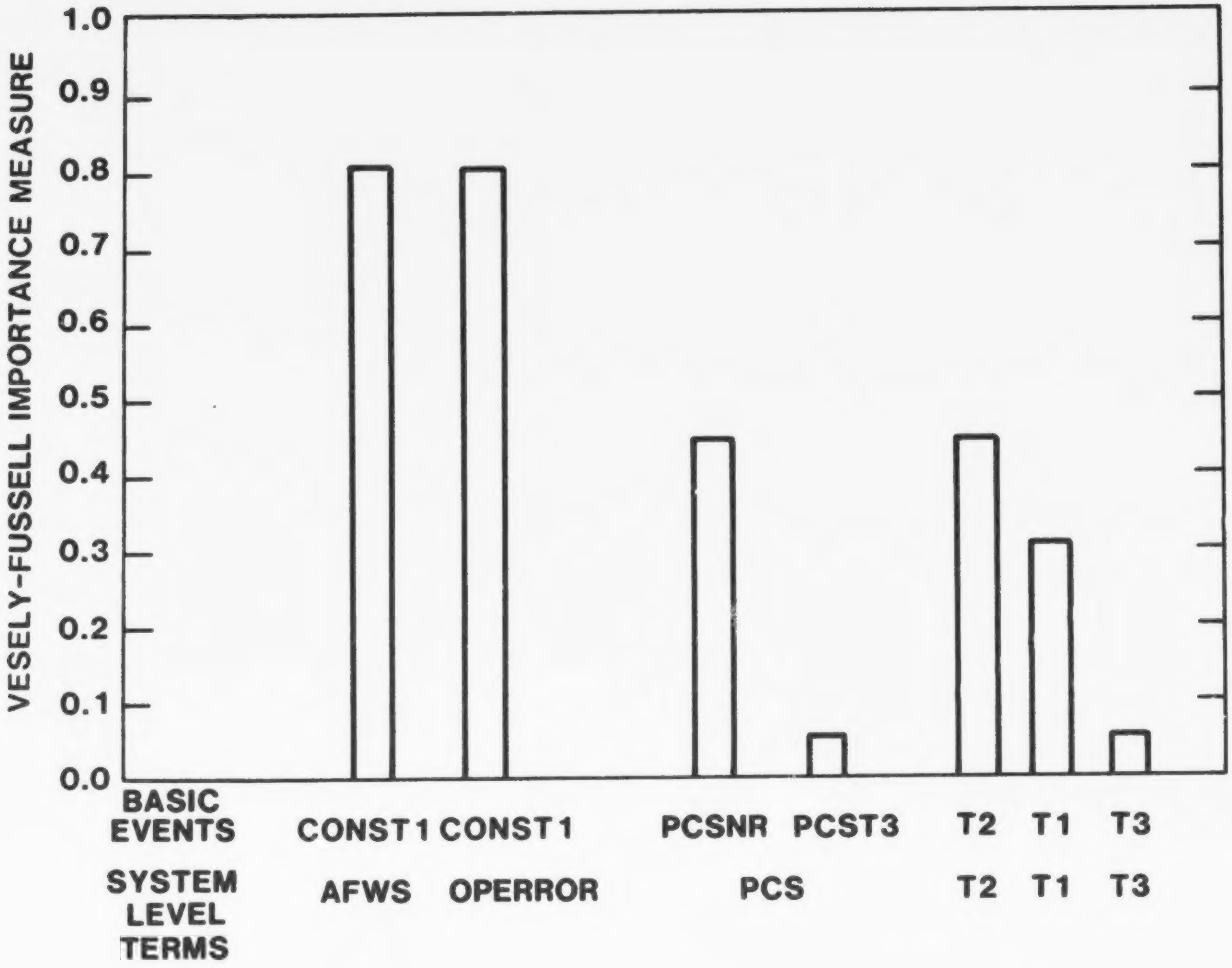

Figure 35. Calvert Cliffs, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

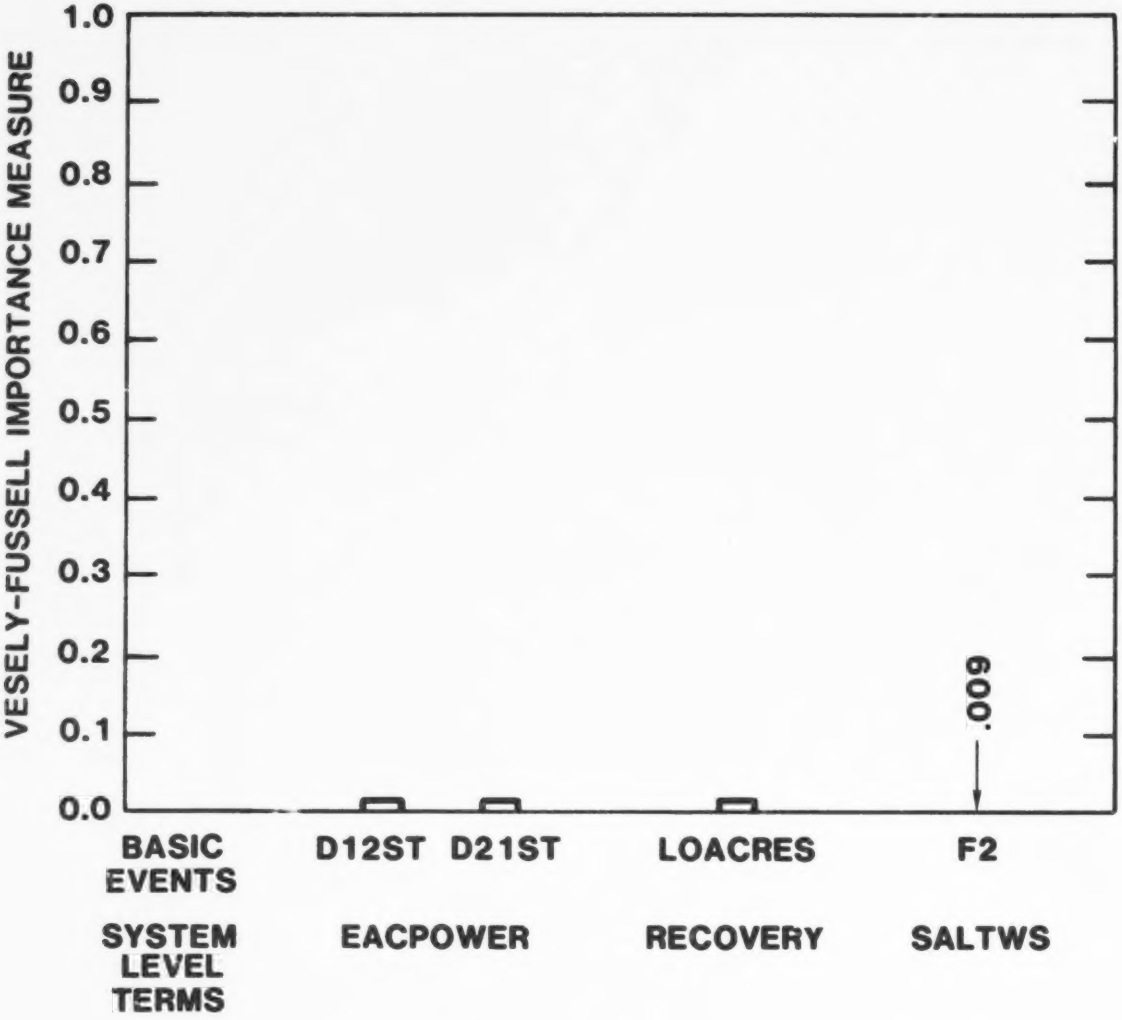

Figure 35. Calvert Cliffs, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

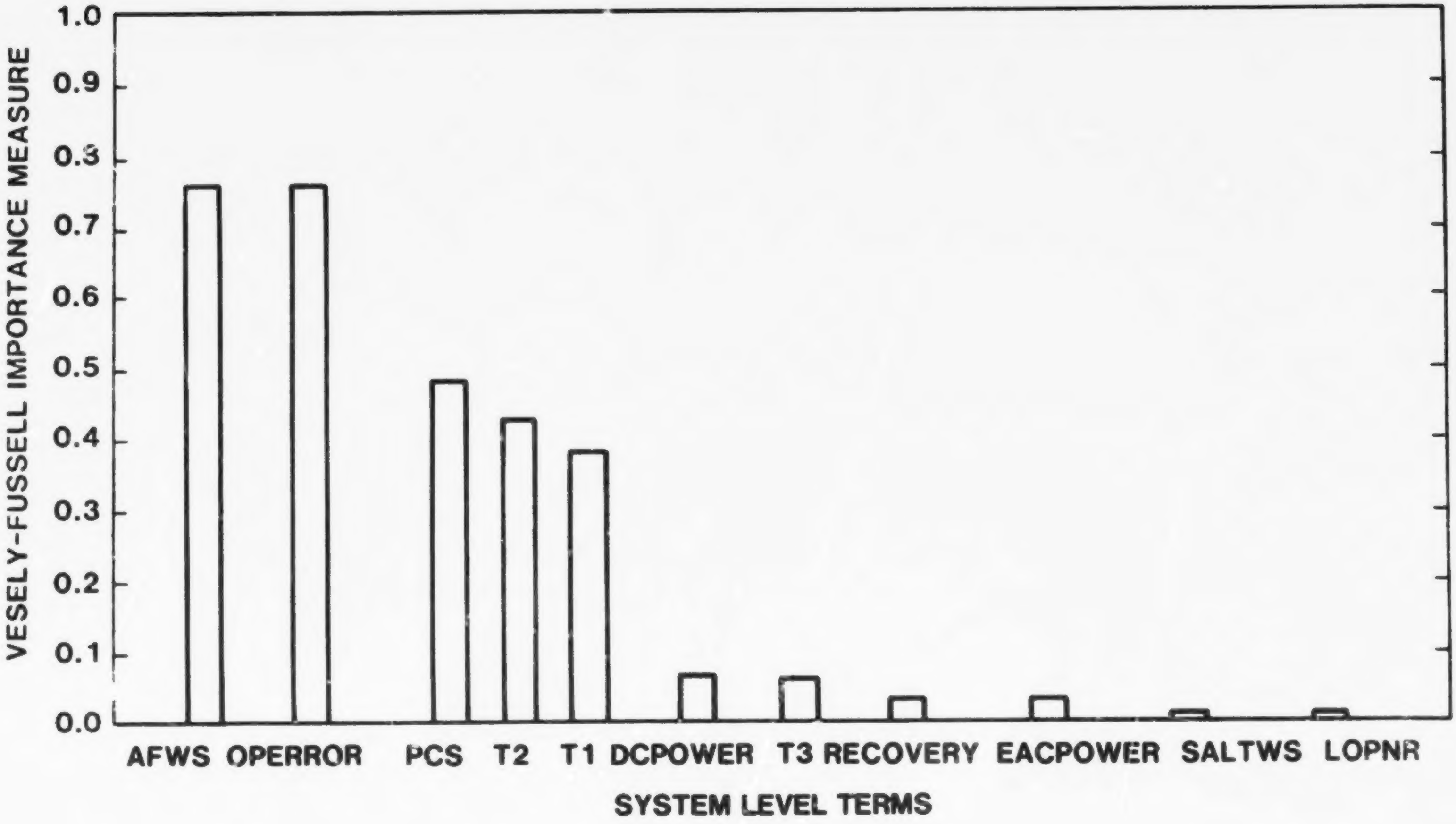

Figure 36. Calvert Cliffs, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

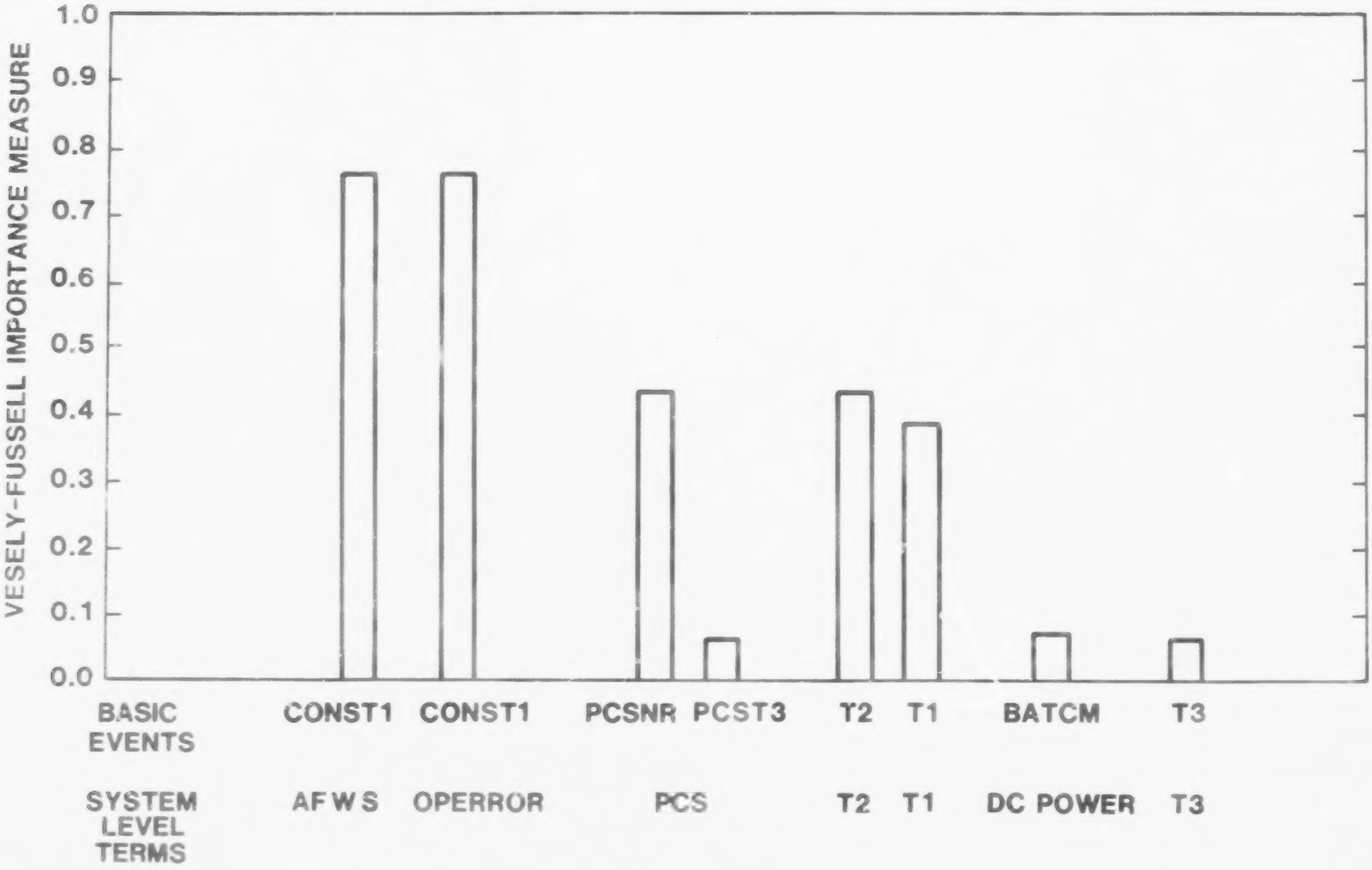

Figure 37. Calvert Cliffs, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

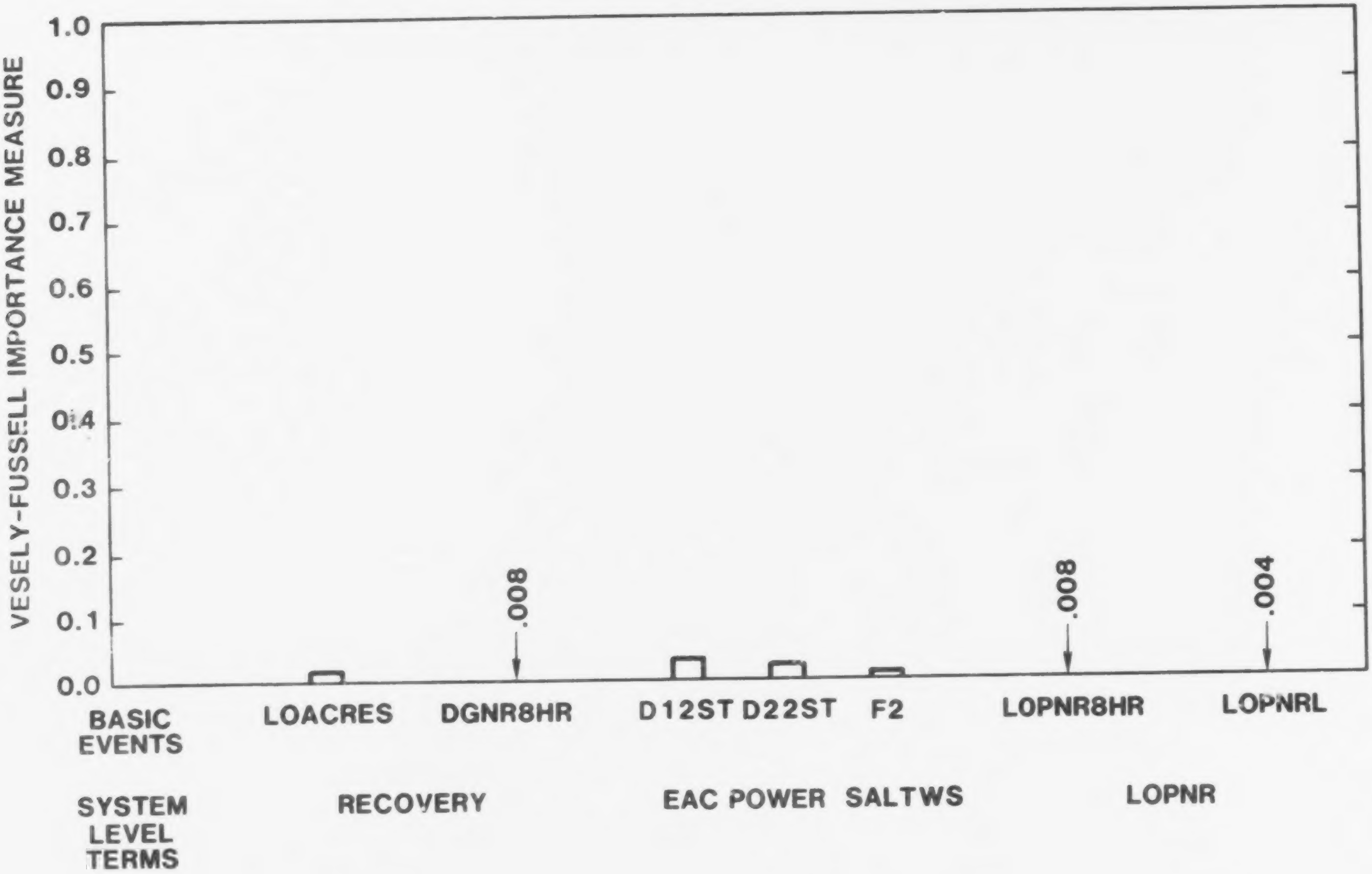

Figure 37. Calvert Cliffs, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

2. Criterion 3; Total Plant Risk

Four sequences are dominant to plant risk. No significant differences exist between the results for Criterion 2 and the results for Criterion 3. Hardware and operator faults in the AFWS still dominate all other events with a significant contribution to plant risk by the PCS faults. However, the inclusion of sequence T_1MLO(O) in the Criterion 3 analysis accounts for a small but significant importance of the DC Power System. This fault, BATCM, is a miscalibration of the battery charger charging rate, which allows the batteries to degrade and fail when needed. This fault is actually a human error, even though in this analysis it is modeled as a DC Power System fault. This event is described in Appendix A2 of the PRA. BATCM is independent of all other system faults and operator actions.

B. Importance of Systems and Components-BWR Results

1. General Electric Plant

Browns Ferry-1
Grand Gulf
Peach Bottom
Limerick
Big Rock Point
Millstone 1

a. Browns Ferry-1

The analysis for Browns Ferry-1 was done using the Interim Reliability Evaluation Program (IREP) study on Browns Ferry. This report was done by EG and G Idaho for the NRC. Sequence fault trees and cut sets were not explicitly listed or shown in the PRA for the dominant sequences. By using the fault tree figures of the dominant sequences in Appendix C of the study (which are not actual fault trees, but simplified sequence logic diagrams), and the system cut sets in Appendix B, sequence cut sets were developed. The resulting cut sets were reduced to minimal cut sets for each sequence. These cut sets are listed in Appendix A9.

1. Criteria 1 and 2; Total Core Melt Frequency

Three sequences are dominant to core melt frequency (Table 76). The frequency of these sequences account for almost 90 percent of the total core melt frequency, yet the cut sets developed for these sequences model only

Table 76

Browns Ferry-1

Plant Vendor	Containment	MWe Rating	PRA Study
General Electric	Mark I	1098	IREP

Dominant Accident Sequences

Sequence	Contribution to Overall Criteria Impact Measures		
	1	2	3
1 $T_uR_BR_A$	.50	.50	.50
2 T_uB	.26	.26	.26
3 $T_pR_BR_A$	.14	.14	.14

Dominant Accident Sequence Release Category Probabilities

Sequence	Reactor Safety Study BWR Release Categories							Sequence Frequency $(R\text{-}yr)^{-1}$
	1	2	3	4	5	6	7	
1 $T_uR_BR_A$	.0001	0.2	0.8					9.7E-5
2 T_uB	.0001	0.2	0.8					5.1E-5
3 $T_pR_BR_A$	.0001	0.2	0.8					2.8E-5

Total Plant Core Melt Frequency 2.0E-4

Table 77

Browns Ferry-1 System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
EACPOWER	Emergency AC Power System	ADL001AR ADL001BR ADL001CR ADL001DR ADL003AR ADL003BR ADL0033RR DG3COMB
EECW	Emergency Equipment Cooling Water System	KCK001NG KCK008NG KCK055SG KCK011
OPERROR	Operator Error	RRA0001D RRB0001D
RCIC	Reactor Core Isolation Cooling System	QCK008UG QCK025UG QCK039UR QPT010UR ORD028UF QS42B40J QVM008UT QVM039UT
RHRS	Residual Heat Removal System	RCK0071G RCK0302G RCK0571G RCK0712G RCK047UG RCK048UG RR3A112 RR3A122
RHRSW	Residual Heat Removal Service Water System	SUPRA SUPRB SUPRC SUPRD
RPS	Reactor Protection System	B M

Table 77 (Cont.)

Browns Ferry
System-Level Grouping of
Boolean Terms From Sequence Fault Trees

System	System Description	Component
T_1	Loss of Offsite Power Induced Transients	TP TPK
T_3	PCS Unavailable Transients	TA TK TU

Table 78

Browns Ferry-1
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
ADL001AR	Diesel Generator Failure to Start	3.0E-2
ADL001BR	Diesel Generator Failure to Start	3.0E-2
ADL001CR	Diesel Generator Failure to Start	3.0E-2
ADL001DR	Diesel Generator Failure to Start	3.0E-2
ADL003AR	Diesel Generator Failure to Start	3.0E-2
ADL003BR	Diesel Generator Failure to Start	3.0E-2
B	Failure to Scram Reactor	3.0E-2
DG3COMB	Electric Power Faults, Combination of Three Diesel Generators Failing.	2.9E-2
KCK001NG	Emergency Equipment Cooling Water (EECW) System, Motor Control Circuit for FCV-67-1	3.2E-3
KCK008NG	EECW, Motor Control Circuit for FCV-67-8	3.2E-3
KCK001SG	EECW, Motor Control Circuit for FCV-67-11	3.2E-3
KCK005SG	EECW, Motor Control Circuit for FCV-67-5	3.2E-3
M	Recirculation Pumps Isolation Cooling System (RCIC)	7.3E-2
QCK008UG	Reactor Core Isolation Cooling System (RCIC) FCV-71-P Control Circuit	3.2E-3
QCK025UG	RCIC, FCV-71-25 Control Circuit	3.2E-3
QCK039UG	RCIC, FCV-71-39 Control Circuit	3.2E-3
QPT010UR	RCIC Pump	3.0E-3
QRD028UF	Turbine Exhaust Line Rupture Disk in RCIC	5.7E-5

Table 78 (Cont.)

Browns Ferry-1
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
QS42B40J	Initiation and Isolation Logic Functional Test SI4.2B-40A in RCIC	1.9E-3
QVM008UT	Steam Stop Valve FCV-71-P in RCIC	1E-3
QVM025UT	Cooling Water Supply Valve FCV-71-25 in RCIC	1E-3
QVM039UT	Discharge Valve FCV-71-39 in RCIC	1E-3
RCK047UG	Control Circuit Fault-No Output, Residual/Heat Removal System (RHR)	3.3E-3
RCK048UG	Control Circuit Fault-No Output, RHR	3.3E-3
RCK071G	Control Circuit Fault-No Output, RHR	3.3E-3
RCK0302G	Control Circuit Fault-No Output, RHR	3.3E-3
RCK0571G	LOOP 1 Inboard Torus Cooling Valve, RHR	3.3E-3
RCK0712G	LOOP 2 Inboard Torus Cooling Valve, RHR	3.3E-3
RRA0001D	Manual Initiation Shutdown Cooling Mode, RHR	1.0E-3
RRB0001D	Manual Initiation Shutdown Cooling Mode, RHR	1.0E-3
RR3A112	Isolation Signal Fault, RHR	4.7E-3
RR3A122	Isolation Signal Fault, RHR	4.7E-3
SUPRA	Residual Heat Removal Service Water System (RHRSW) Interface Fault	2.0E-2
SUPRB	RHRSW Interface Fault	2.0E-2
SUPRC	RHRSW Interface Fault	2.0E-2

Table 78 (Cont.)

Browns Ferry-1
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
SUPRD	RHRSW Interface Fault	2.0E-2
TA	Transient With PCS Available	1.68/R-yr
TK	Transient Induced LOCA	1.63-1/R-yr
TP	Loss of Offsite Power (LOSP)	3.0E-2
TPK	LOSP - Induced LOCA	1.7E-3/R-yr
TU	PCS Unavailable After Transient	1.7/R-yr

Table 79

Browns Ferry-1

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
T_3	.26	.86	.66
RPS	.26	.86	.65
T_1	.14	.74	.34
EACPOWER	.13	.73	.33
RHRS	.006	.61	.02
OPERROR	.004	.60	.01
RHRSW	.002	.60	.005
EECW	5.3E-5	.60	.0001

Table 80

Browns Ferry-1

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
T_3	TU	.26	.86	.66
RPS	B	.26	.86	.65
T_1	TP	.14	.74	.34
EACPOWER	DG3COMB	.13	.73	.33
	ADL001DR	.0002	.01	.0004
	ADL003AR	.0001	.60	.0003
	ADL001CR	.0001	.60	.0003
	ADL003BR	.0001	.60	.0003
	ADL001AR	4.1E-5	.60	.0001
	ADL001BR	8.3E-5	.60	.0002
	ADL001AR	4.1E-5	.60	.0001
	ADL003RR	4.4E-6	.60	1.1E-5
RHRS	RR3A112	.002	1.61	.004
	RR3A122	.002	.61	.004
	RCK048UG	.001	.60	.003
	RCK047UG	.001	.60	.003
	QCK0302G	.0005	.60	.001
	RCK0571G	.0005	.60	.001
	RCK0712G	.0005	.60	.001
	RCK0071G	.0005	.60	.001
OPERROR	RRB0001D	.004	.60	.01
	RRA0001D	.003	.60	.0009
RHRSW	SUPRA	.0005	.60	.001
	SUPRB	.0005	.60	.001
	SUPRC	.0005	.60	.001
	SUPRD	.0005	.60	.001
EECW	KCK005SG	1.3E-5	.60	3.3E-5
	KCK001SG	1.3E-5	.60	3.3E-5
	KCK008NG	1.3E-5	.60	3.3E-5
	KCK001NG	1.3E-5	.60	3.3E-5

Table 81

Browns Ferry-1

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
T_3	.26	.86	.66
RPS	.26	.86	.65
T_1	.14	.74	.34
EACPOWER	.13	.73	.33
RHRS	.006	.61	.02
OPERROR	.004	.60	.01
AHRSW	.002	.60	.005
EECW	5.3E-5	.60	.0001

Table 82

Browns Ferry-1

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk
Criterion 3
(Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
T_3	TU	.26	.86	.66
RPS	B	.26	.86	.65
T_1	TP	.14	.74	.34
EACPOWER	DG3COMB	.13	.73	.33
	ADL001DR	.0002	.60	.0004
	ADL00AD	.0001	.60	.0003
	ADL001CR	.0001	.60	.0003
	ADL003BR	.0001	.60	.0003
	ADL001BR	8.3E-5	.60	.0002
	ADL001AR	4.1E-5	.60	.0001
	ADL003RR	4.5E-6	.60	1.1E-5
RHRS	RR3A112	.002	.60	.004
	RR3A122	.002	.60	.004
	RCK048UG	.001	.60	.003
	RCK047UG	.001	.60	.003
	RCK0302G	.0005	.60	.001
	RCK0572G	.0005	.60	.001
	RCK07126	.0005	.60	.001
	RCK0071G	.0005	.60	.001
OPERROR	RRB0001D	.004	.60	.01
	RRA000D	.0004	.60	.0009
RHRSW	SUPRA	.0005	.60	.001
	SUPRB	.0005	.60	.001
	SUPRC	.0005	.60	.001
	SUPRD	.0005	.60	.001
EECW	KCK005SG	1.3E-5	.60	3.3E-5
	KCK011SG	1.3E-5	.60	3.3E-5
	KCK008NG	1.3E-5	.60	3.3E-5
	KCK001NG	1.3E-5	.60	3.3E-5

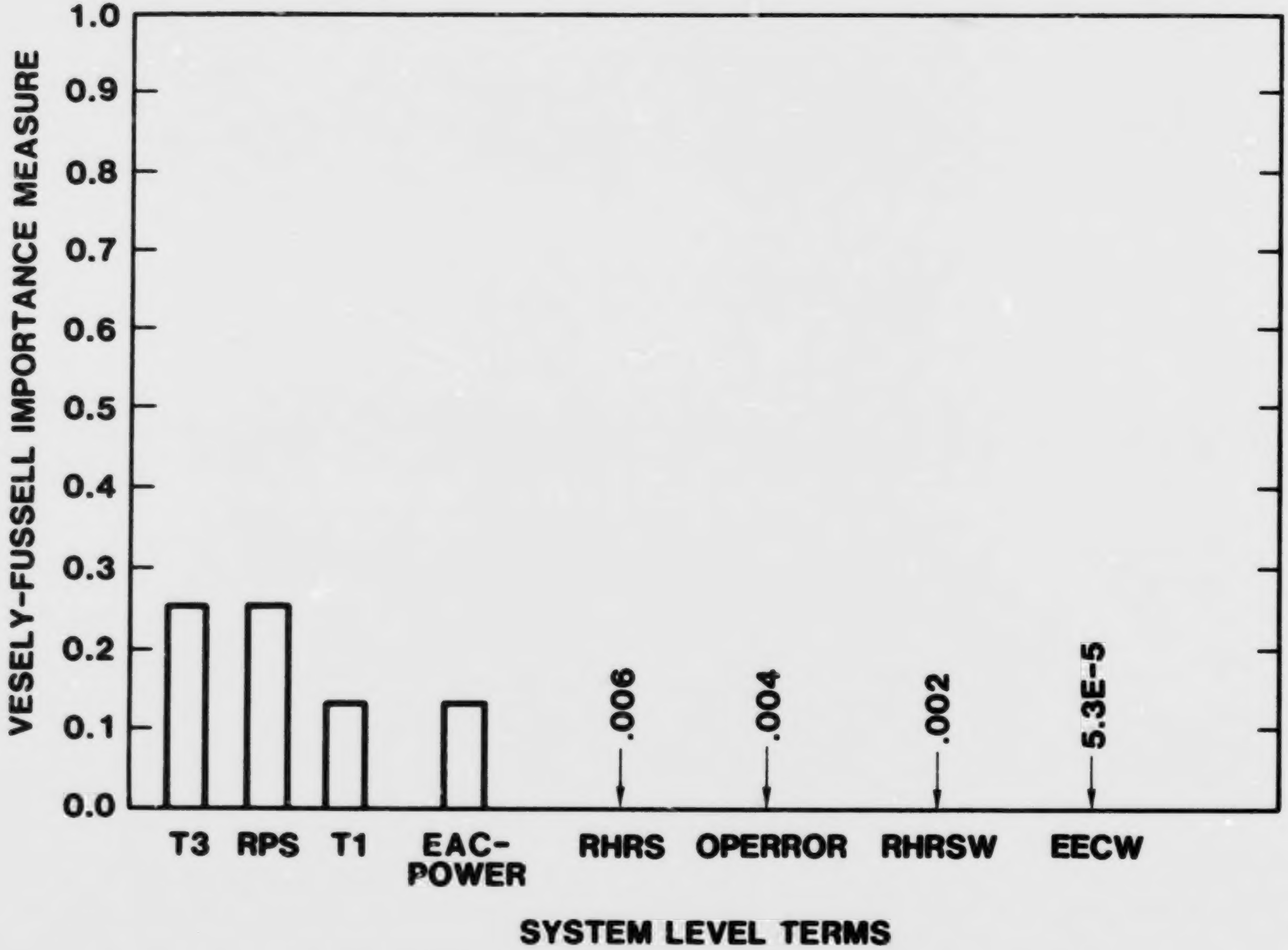

Figure 38. Browns Ferry-1, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

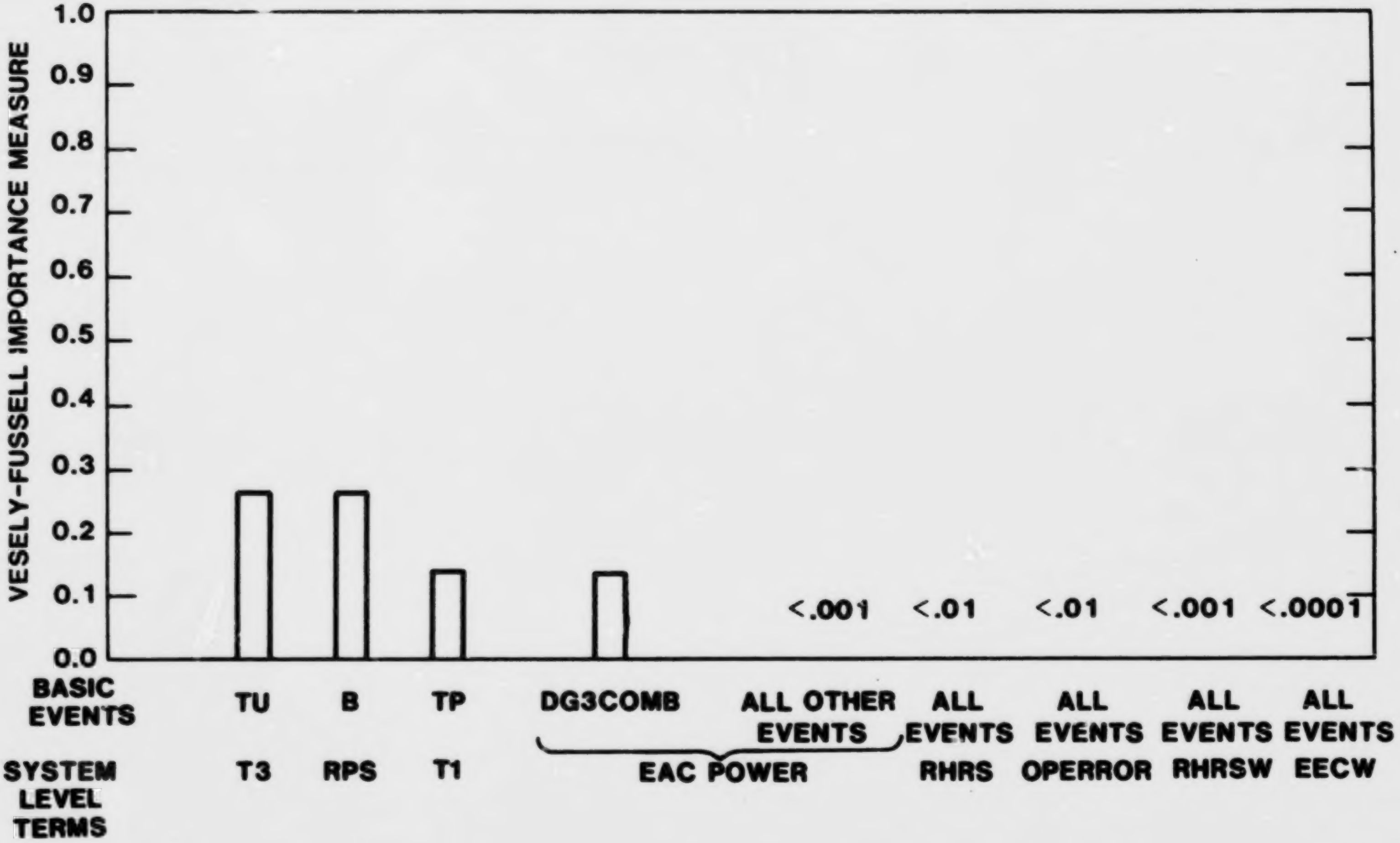

Figure 39. Browns Ferry-1, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

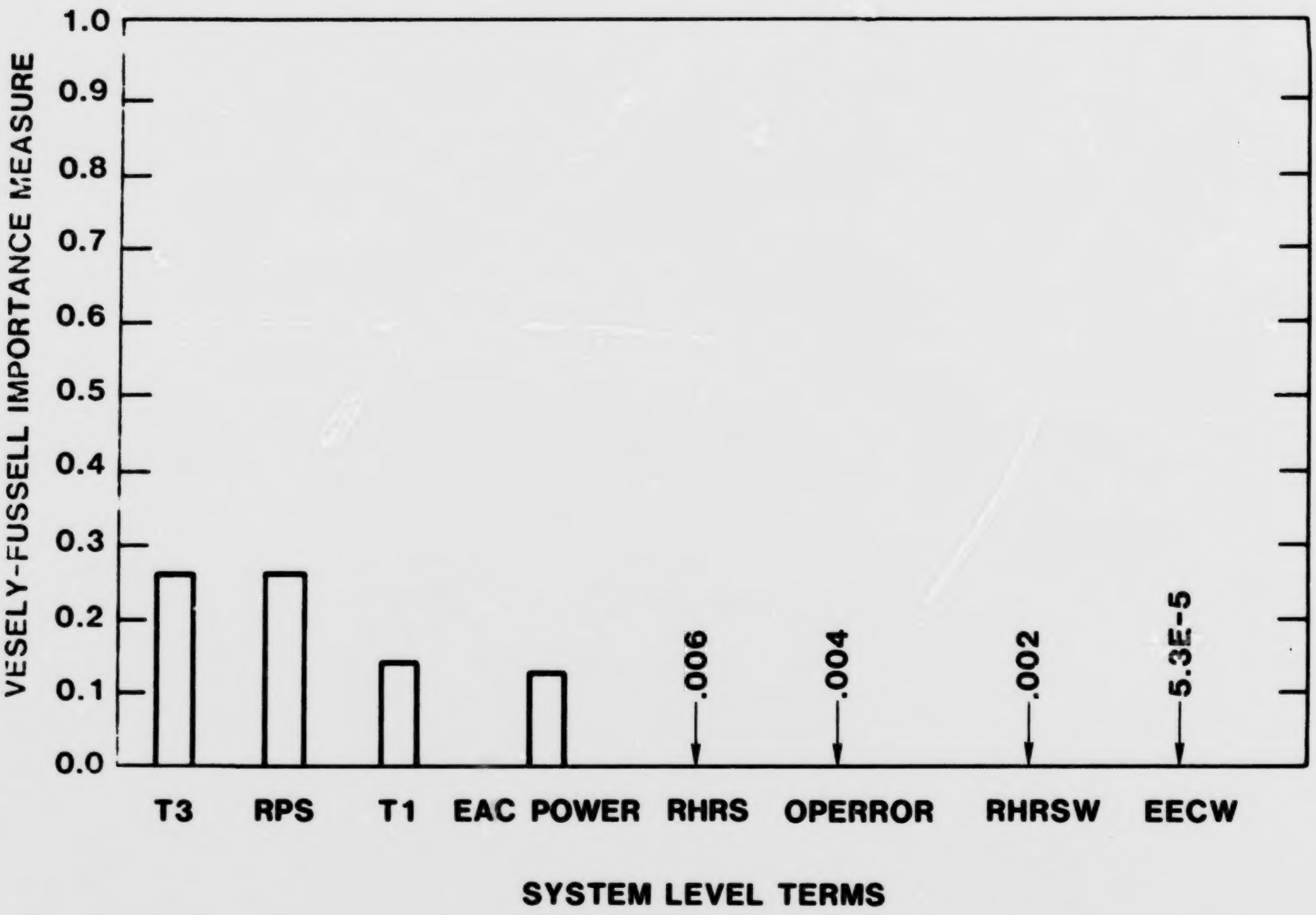

Figure 40. Browns Ferry-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

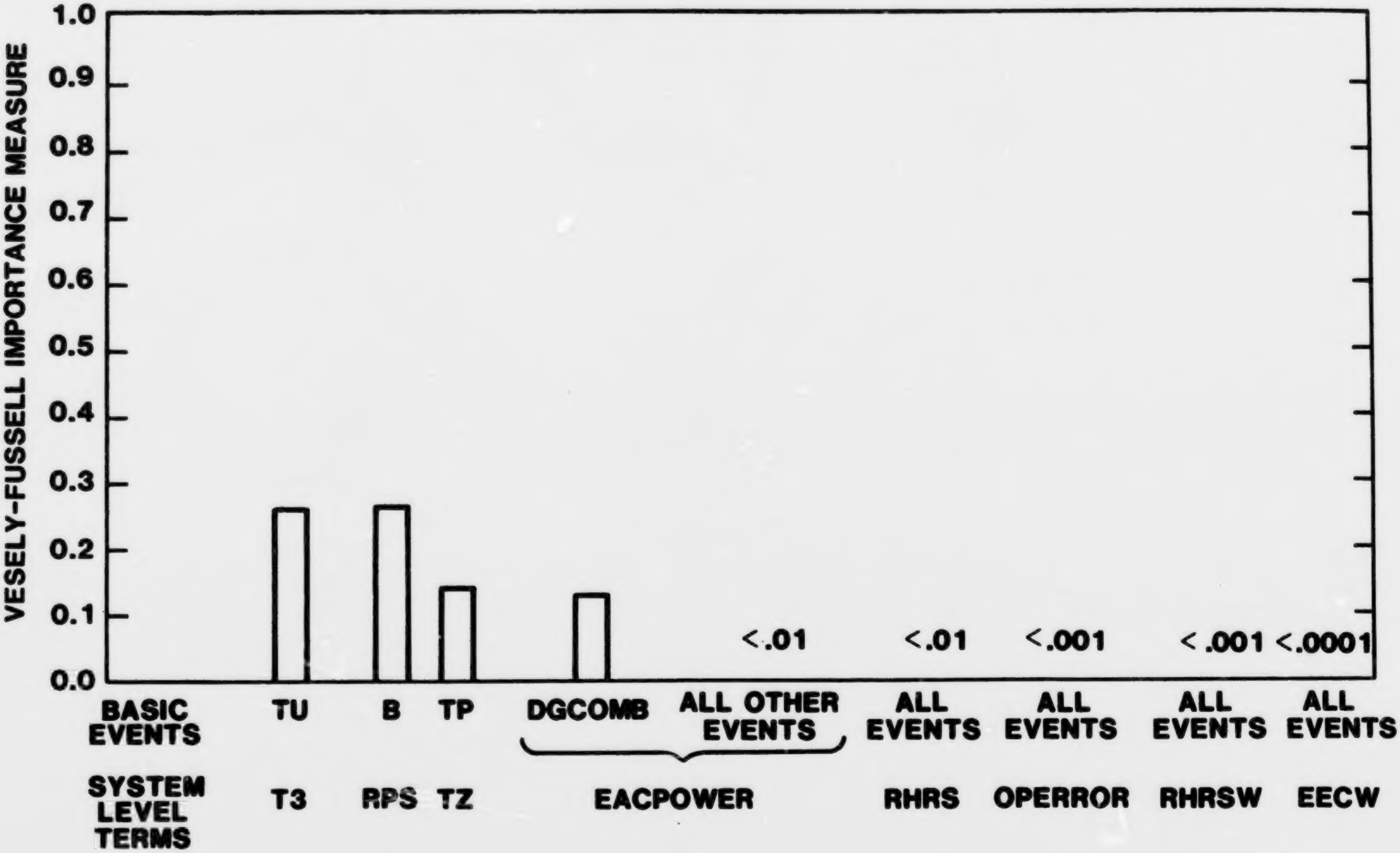

Figure 41. Browns Ferry-1, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

40 percent of the core melt frequency. This is caused by limited reporting of system cut sets in the PRA.

All of the dominant sequences in the Browns Ferry PRA are transient sequences, so no LOCA events appear in the importance results. The system level results (Table 79, Figure 38) show only two systems with significant importance, the Reactor Protection System (RPS), and emergency AC Power (EACPOWER). Each system is dominated by one event. The RPS fault is an event tree level event, event B in the PRA. The cut set and sequence (T_uB) that event B appears in are simply two events, the transient T_u and B. The unavailability for event B is taken from NUREG-0460. The dominant fault of the emergency AC power system, DG3COMB, is taken from Figure C-15 of the PRA in the discussion of sequence $T_PR_BR_A$ in the PRA. This fault is some combination of the diesel generators failing. However, no description or quantification of this event could be located anywhere in the PRA.

The insignificant importance of the RHRS should be viewed with extreme suspicion. The two dominant sequences which involve failure of the RHRS ($T_UR_BR_A$) and ($T_PR_BR_A$) account for over 60 percent of core melt frequency. The unavailabilities for the RHRS system shown for these sequences on Figures C-15 and C-19 in the PRA show that RHRS is, at least, very significant to each sequence. RHRS should therefore be important to core melt frequency. However, after developing the system cut sets for the RHRS from the tables in the Browns Ferry PRA Appendix B and reducing to minimal cut sets, the results do not even approximate the unavailabilities for RHRS shown on the PRA's Figures C-15 and C-19. So little information is reported in the systems cut set tables in Appendix B, that the system unavailability in the fault tree figure in Appendix C can not be reproduced by generating sequence cut sets.

2. Criterion 3; Total Plant Risk

The results for Criterion 3 are virtually the same as for Criterion 2, since the same sequences dominate both Criteria, and the three sequences have the same release category probabilities (Table 76). These results are subject to the same uncertainties and limitations as those for Criteria 1 and 2.

b. Grand Gulf-1

The analysis for Grand Gulf was done using the Reactor Safety Study Methodology Applications Program (RSSMAP) study on Grand Gulf.[19] Cut sets for the dominant sequences were taken directly from Appendix D of the PRA. Basic events descriptions (Table 85) were taken from the various appendices on plant systems and the discussions of dominant sequences. The cut sets of the dominant sequences which where analyzed model approximately 83 percent of core melt frequency, and 88 percent of plant risk. These cut sets are shown in Appendix All.

1. Criteria 1 and 2; Total Core Melt Frequency

Five sequences contribute more than 5 percent to core melt frequency, four transient sequences and one LOCA sequence (Table 83). The system level importances terms are clearly dominated by three plant system level terms, the Standby Service Water System (SSWS), Recovery actions by plant personnel (RECOVERY), and the Power Conversion System (PCS). The seven other systems show significant contributions to core melt frequency as well (Table 86, Figure 42).

The SSWS is dominated by six basic events (Table 87, Figure 43). The SSWS supplies cooling to the RHRS heat exchanges. Four of the dominant sequences involve failure of the RHRS to remove heat from the suppression pool or the containment. Inspection of the system level cut sets in Appendix All shows that SSWS failures are in most of the cut sets of these sequences. Only a few cut sets contain RHRS failures. So the large importance of SSWS reflects the dependences of RHRS success upon SSWS success. The two most basic events of the SSWS failures (SSA, SSB) are described in detail in Appendix B12 of the PRA. The next four dominant faults (VGB1, VGA1, VGA2, VGB2) are described in Appendix B10 of the PRA. These faults are valve and pump failures in the SSWS A and B trains. Operator errors, test and maintenance faults, and hardware faults have been lumped together in defining these events. So the actual amount of importance due to human or hardware faults cannot be determined from the importance calculations.

The system level term RECOVERY is composed of two basic terms, RECOVERY1 and RECOVERY2. In the PRA, there is only one recovery term, but for this analysis a distinction was made for those sequences which allow 28 hours for recovery actions to take place (RECOVERY1), and those sequences which allow 30 hours for recovery actions to take place (RECOVERY2). Recovery is described in a general way for each sequence.

The system level PCS failure is an event tree level event as well as a basic event. This event is discussed in Appendix A.2 of the PRA. WASH-1400 values were used for PCS unavailability. The importance of the PCS in this analysis represents the PCS failure contribution due to T_{23} transient. In T_1 transient cut sets, PCS unavailability is modeled by events LOPNRE and LOPNRL which are basic events treated as system level terms as well (Table 84).

These events represent failure to recover offsite power early in the sequence and late in the sequence. These events are also discussed in Appendix A2 of the PRA under the PCS.

The emergency AC power system (EACPOWER) is dominated by the failure of diesel generators 1 and 2 (DIESEL1, DIESEL2). These events are discussed in Appendix A2 of the PRA. The third diesel generator, DIESEL3, is dedicated to just the High Pressure Core Spray System, which does not appear in any of the dominant sequences analyzed in this study.

The RHRS dominant faults are a combination of maintenance, control circuit, and hardware faults in valves and pumps of RHRS trains A (fault LA2) and B (fault LB2) which fail flow from the suppression pool and recirculation system to the RHR heat exchangers.

In this analysis, the system level event Q, failure of a safety/relief valve to reseat, is called event P in the PRA and at the basic event level. It is labled Q at the system level to be consistant with labeling of this event in other PRAs. This event is discussed in Appendix A2 under event P. The value of this event is taken from WASH-1400.

The Reactor Core Isolation Cooling System (RCICS) main fault is basic event R. This event is discussed in Appendix A6 of the PRA. It is a combination of hardware, human, and maintenance faults in RCICS valves, pumps, and the RCICS turbine.

The failure of the Reactor Protection System (RPS), is a event tree level event in the PRA (event C). This is failure to scram the reactor. The RPS is described in Appendix A1 of the PRA. Its unavailability is based on WASH-1400 data.

2. Criterion 3; Total Plant Risk

The results for Criterion 3 are almost exactly the same as for Criterion 2, as the same sequences dominate both Criteria, and the sequences all have essentially the same category release probabilities (Table 83). The results for Criterion 3 are listed on Tables 88 and 89, and shown on Figures 44 and 45.

Table 83

Grand Gulf

Plant Vendor	Containment	MWe Rating	PRA Study
General Electric	Mark III	1250	RSSMAP

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures 1	2	3
1	$T_{23}PQI$	.10	.10	.11
2	SI	.12	.12	.13
3	T_1QW	.17	.17	.18
4	$T_{23}QW$	.32	.32	.34
5	$T_{23}C$	.15	.15	.16

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study BWR Release Categories 1	2	3	4	5	Sequence Frequency $(R\text{-}yr)^{-1}$
1	$T_{23}PQI$	.01	1.0				3.7E-6
2	SI	.01	1.0				4.6E-6
3	T_1QW		1.0				6.2E-6
4	$T_{23}QW$		1.0				1.2E-5
5	$T_{23}C$		1.0				5.4E-6

Total Plant Core Melt Frequency 3.7E-5

Table 84

Grand Gulf
System-Level Grouping of
Boolean Terms From Sequence Fault Trees

System	System Description	Component
DCPOWER	DC Power System	BATA BATB
EACPOWER	Emergency AC Power	DIESEL 1 DIESEL 2 DIESEL 3
HPCS	High Pressure Core Spray	H HACT
LPCS	Low Pressure Core Spray	L LRACT
LPCIS	Low Pressure Coolant Injection System	BCACT LB1 LB2 LC LRACT
LOPNRE	Loss of Offsite Power Not Recovered Early	LOPNRE
LOPNRL	Loss of Offsite Power Not Recovered Late	LOPNRL
OPERROR	Operator Error	OP
PCS	Power Conversion System	Q
Q	Failure of Safety/Relief Valve to Reset	P
RCICS	Reactor Core Isolation Cooling System	R RACT
RECOVERY	Failure to Restore of Maintenance or Test Faults	RECOVERY1 RECOVERY2

Table 84 (Cont.)

Grand Gulf
System-Level Grouping of
Boolean Terms From Sequence Fault Trees

System	System Description	Component
RHRS	Residual Heat Removal System	BCACT LA2 LB1 LB2 LRACT PA27 PB27 SCVA SCVB
RPS	Reactor Protection System	C
S	Small LOCA	S
SPMS	Suppression Pool Makeup System	SA SAACC SB SBACC
SSWS	Standby Service Water System	SAC SBC SCC SSA SSB SSC VGA1 VGA2 VGB1 VGB2 V1 V2 V3
T_1	Transient Resulting in Loss of Offsite Power	T_1
T_{23}	Transients Other Than LOP but Which Require Reactor Shutdown	T_{23}

Table 85

Grand Gulf
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
BATA	Battery A Fails on Demand	1E-3
BATB	Battery B Fails on Demand	1E-3
BCACT	Initiating Circuit Residual Heat Removal System (RHR) Loops B and C	1.2E-3
C	Failure to Achieve Reactor Subcriticality for T_{23} Initiating Events	7.7E-7
DIESEL 1	Diesel Generator 1 Fails	3.6E-2
DIESEL 2	Diesel Generator 2 Fails	3.6E-2
DIESEL 3	Diesel Generator 3 Fails	3.6E-2
H	High Pressure Core Spray System (HPCS) Valve Malfunctions	2.1E-2
HACT	Failure of HPCS Actuating Circuit	1.2E-3
L	Low Pressure Codent Injection System (LPCIS) Hardware Failure	2.1E-2
LA2	LPCIS Hardware Failure Loop A	1.4E-2
LB1	LPCIS Hardware Failure Loop B	1.3E-2
LB2	LPCIS Hardware Failure Loop B	1.4E-2
LC	LPCIS Hardware Failure Loop C	2.2E-2
LOPNRE	Failure to Recover Offsite Power in One Half Hour	2E-1
LOPNRL	Failure to Recover Offsite Power in 27 Hours Given LOPNRE	1E-1
LRACT	Initiating Circuit RHR Loop A and LPCS	1.2E-3
OP	Operator Failure to Manually Actuate the Automatic Depressuration System (ADS)	1.5E-3
P	Failure of Safety/Relief Valve to Reseat	.1

Table 85 (Cont.)

Grand Gulf
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
PA27	RHR Pump A Fails to Continue Running for About 30 Hours	8.1E-4
PB27	RHR Pump B Fails to Continue Running for About 30 Hours	8.1E-4
Q	Failure of Power Conversion System (PCS) to Remove Heat Within About 28 Hours	7E-3
R	Reactor Core Isolation Cooling System (RCIC) Hardware Failures	5.1E-2
RACT	Failure of RCIC Actuating Circuit	1.2E-3
RECOVERY1	Failure to Restore Maintenance or Test Faults or Take Other Corrective Action Within 28 Hours	.23
RECOVERY2	Failure to Restore Maintenance or Test Faults or Take Other Corrective Action Within 30 Hours	.21
S	A Small Loss of Coolant Accident (Less Than One Square Foot)	1.4E-3
SA	Suppression Pool Makeup System MORV Failures, Train A	1.4E-2
SAACC	SPMS Actuation and Control Circuit A	1.2E-3
SAC	SSWS Actuation and Control Circuit Loop A	1.2E-3
SB	Suppression Pool Makeup System MOV Failures, Train B	1.4E-2
SBACC	SPMS Actuation and Control Circuit B	1.2E-3
SBC	SSWS Actuation and Control Circuit Loop B	1.2E-3
SCC	Actuation and Control Circuit Loop C	1.2E-3
SCVA	Hardware Failure in RHRS, Train A	3.2E-2

Table 85 (Cont.)

Grand Gulf
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
SCVB	Hardware Failure RHRS, Train B	3.2E-2
SSA	Standby Service Water System Hardware Failures, Loop A	2.1E-2
SSB	Standby Service Water System Hardware Failures, Loop B	2.1E-2
SSC	Standby Service Water System Hardware Failures, Loop C	1.4E-2
T_1	Transient Resulting in Loss of Offsite Power	.21/R-yr
T_{23}	Transient Other Than Loss of Offsite Power Which Requires Reactor Shutdown	7/R-yr
VGA1	Motor Operated Valves Failures in SSWS System	1.5E-2
VGA2	Motor Operated Valves Failures in SSWS System	2.4E-2
VGB1	Motor Operated Valves Failures in SSWS System	1.5E-2
VGB2	Motor Operated Valves Failures in SSWS System	2.4E-2
V1	Standby Service Water Valves F018A-A and F023A Unavailable	8.0E-3
V2	Standby Service Water Valves F018B-B and F023B Unavailable	8.0E-3
V3	Standby Service Water Valves F185A, and F185B, F186A, and F186B	3.3E-3

Table 86

Grand Gulf

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action or Initiating Event	Importance	Worst Case Importance	Proportional Importance
SSWS	.64	.81	.77
RECOVERY	.56	.73	.67
T_{23}	.55	.72	.66
PCS	.40	.57	.48
T_1	.15	.33	.19
LOPNRE	.15	.33	.19
LOPNRL	.15	.33	.19
RPS	.15	.32	.18
S	.13	.30	.15
EACPOWER	.11	.28	.13
RHRS	.10	.28	.12
Q	.10	.27	.12
RCICS	.05	.23	.06
SPMS	.007	.18	.008
DCPOWER	.002	.17	.002

Table 87

Grand Gulf

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
SSWS	SSA	.25	.43	.30
	SSB	.25	.43	.30
	VGB1	.15	.33	.18
	VGA1	.15	.33	.18
	VGA2	.11	.30	.14
	VGB2	.11	.30	.14
	SAC	.02	.20	.02
	SBC	.02	.20	.02
	V1	.01	.20	.02
	V2	.01	.20	.02
RECOVERY	RECOVERY2	.45	.63	.55
	RECOVERY1	.10	.28	.12
T_{23}	T_{23}	.54	.72	.66
PCS	Q	.40	.57	.48
T_1	T_1	.15	.33	.19
LOPNRE	LOPNRE	.15	.33	.19
LOPNRL	LOPNRL	.15	.33	.19
RPS	C	.15	.33	.19
S	S	.13	.30	.15
EACPOWER	DIESEL1	.07	.25	.08
	DIESEL2	.07	.25	.08

Table 87 (Cont.)

Grand Gulf

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
RHRS	LA2	.04	.22	.05
	LB2	.04	.22	.05
	BCACT	.006	.19	.007
	LRACT	.006	.19	.007
	SCVA	.005	.18	.006
	SCVB	.005	.18	.006
	PB27	.002	.18	.003
	PA27	.002	.18	.003
Q	P	.10	.27	.12
RCICS	R	.05	.23	.06
SPMS	SA	.005	.18	.007
	SB	.005	.18	.007
	SAACC	.001	.18	.001
	SBACC	.001	.18	.001
DCPOWER	BATA	.0008	.18	.001
	BATB	.0008	.18	.001

Table 88

Grand Gulf

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
SSWS	.67	.80	.77
RECOVERY	.59	.71	.67
T_{23}	.58	.70	.66
PCS	.42	.55	.48
T_1	.16	.29	.19
LOPNRE	.16	.29	.19
LOPNRL	.16	.29	.19
RPS	.16	.28	.18
S	.13	.26	.15
EACPOWER	.12	.24	.13
RHRS	.11	.23	.13
Q	.11	.23	.12
RCICS	.06	.18	.06
SPMS	.007	.13	.008
DCPOWER	.002	.13	.002

Table 89

Grand Gulf

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
SSWS	SSA	.26	.39	.30
	SSB	.26	.39	.30
	VGB1	.16	.29	.18
	VGA1	.16	.29	.18
	VGB2	.13	.25	.14
	VGA2	.13	.25	.14
	SAC	.02	.14	.02
	SBC	.02	.14	.02
	V1	.02	.14	.02
	V2	.02	.14	.02
RECOVERY	RECOVERY2	.48	.60	.55
	RECOVERY1	.11	.23	.12
T_{23}	T_{23}	.58	.70	.66
PCS	Q	.42	.55	.48
T_1	T_1	.16	.29	.19
LOPNRE	LOPNRE	.16	.29	.19
LOPNRL	LOPNRL	.16	.29	.19
RPS	C	.16	.28	.18
S	S	.13	.26	.15
EAPOWER	DIESEL1	.07	.20	.08
	DIESEL2	.07	.20	.08
RHRS	LA2	.04	.17	.05
	LB2	.04	.17	.05
	BCACT	.007	.13	.007
	2RACT	.007	.13	.007
	SCVB	.006	.13	.006
	SCVA	.006	.13	.006
	PB27	.003	.13	.003

Table 89 (Cont.)

Grand Gulf

Contribution of Basic Events, Grouped by Front Line
and Support System, to Total Plant Risk:
Criterion 3
(Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
	PA27	.003	.13	.003
Q	P	.11	.23	.12
RCICS	R	.06	.18	.06
SPMS	SA	.006	.13	.007
	SB	.006	.13	.007
	SAACC	.001	.13	.001
	SBACC	.001	.13	.001
DCPOWER	BATA	.0009	.12	.001
	BATB	.0009	.12	.001

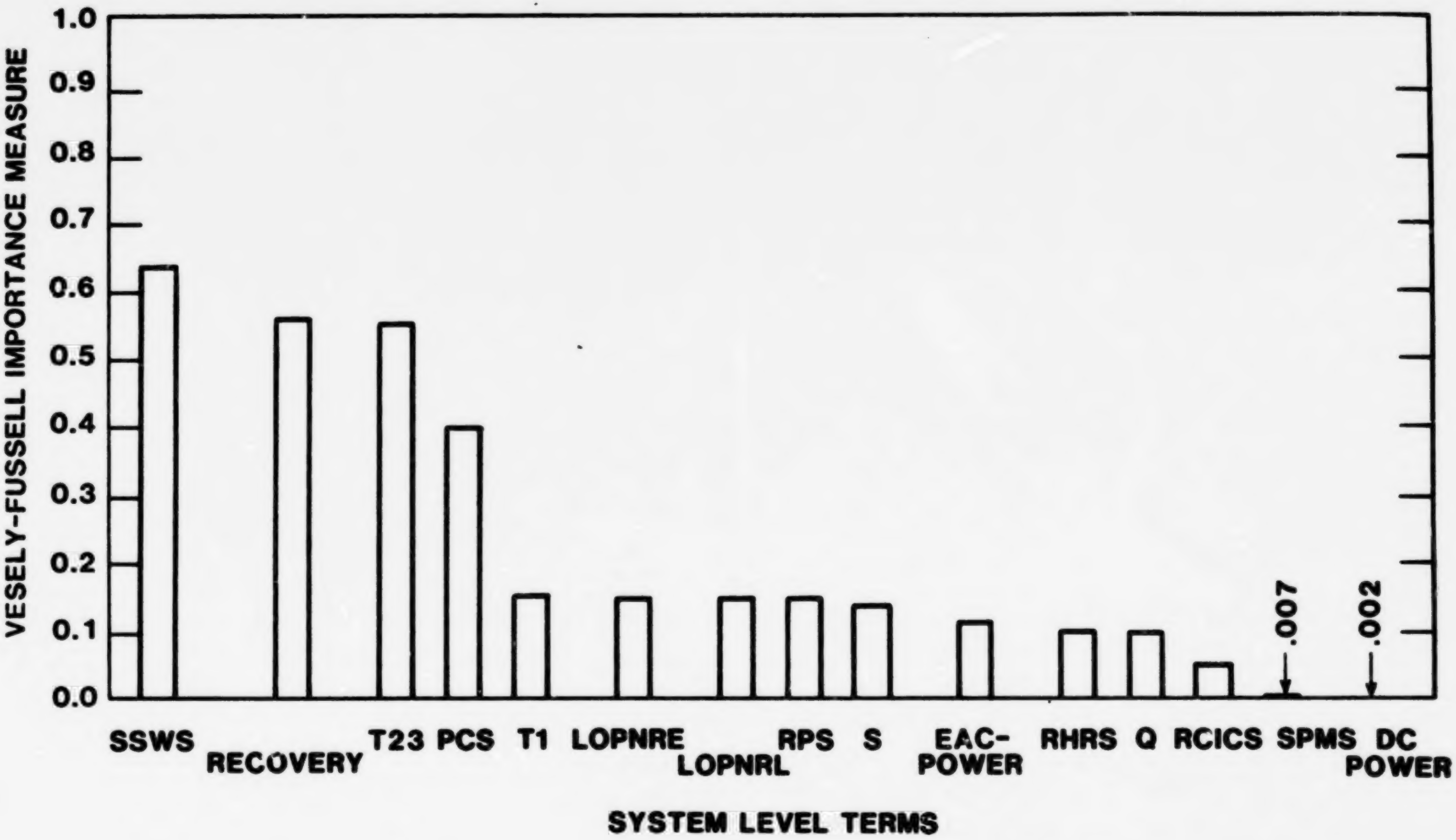

Figure 42. Grand Gulf, Importance of Front Line and Support Systems, to Total Core Melt Frequency: Criteria 1 and 2

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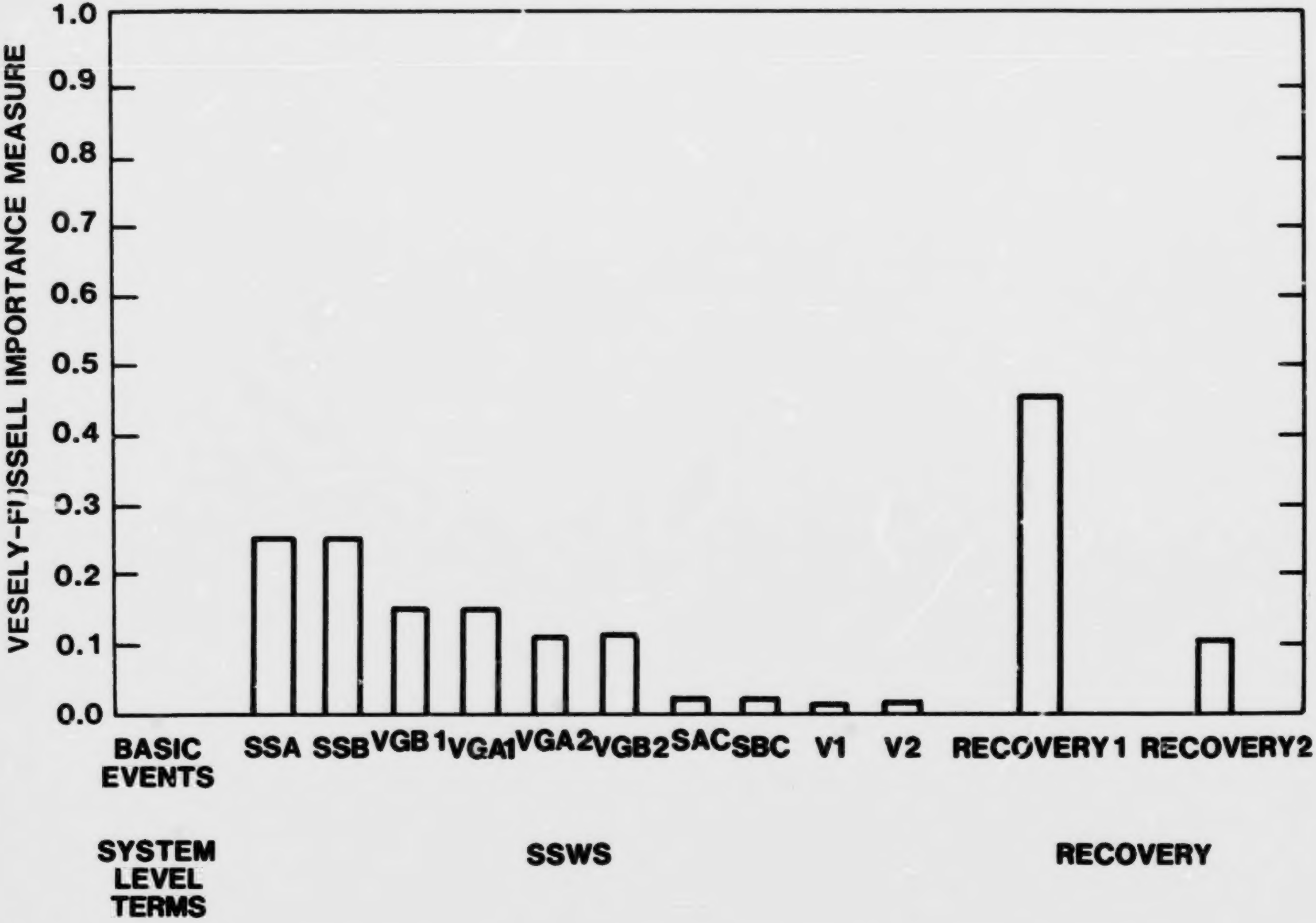

Figure 43. Grand Gulf, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

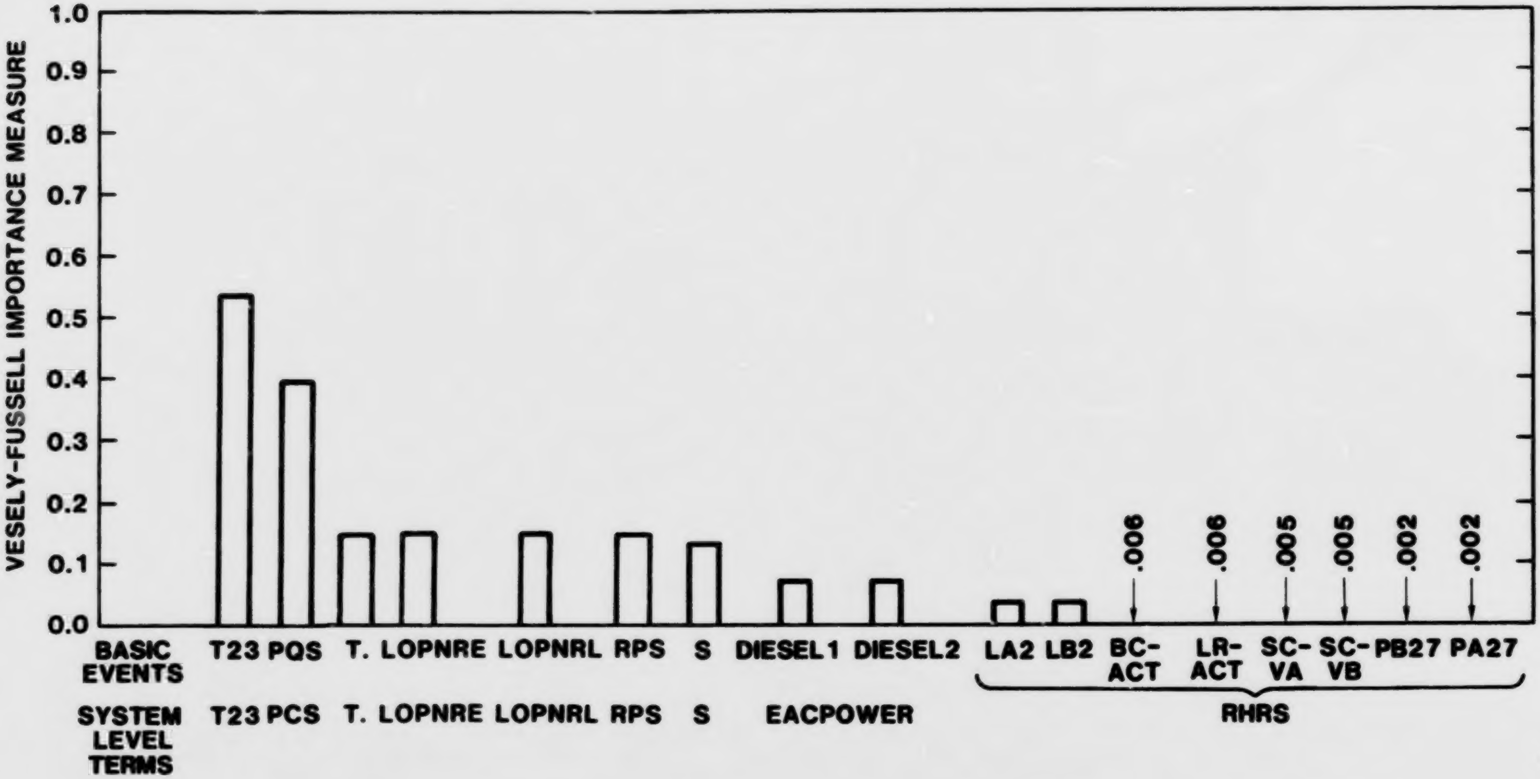

Figure 43. Grand Gulf, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

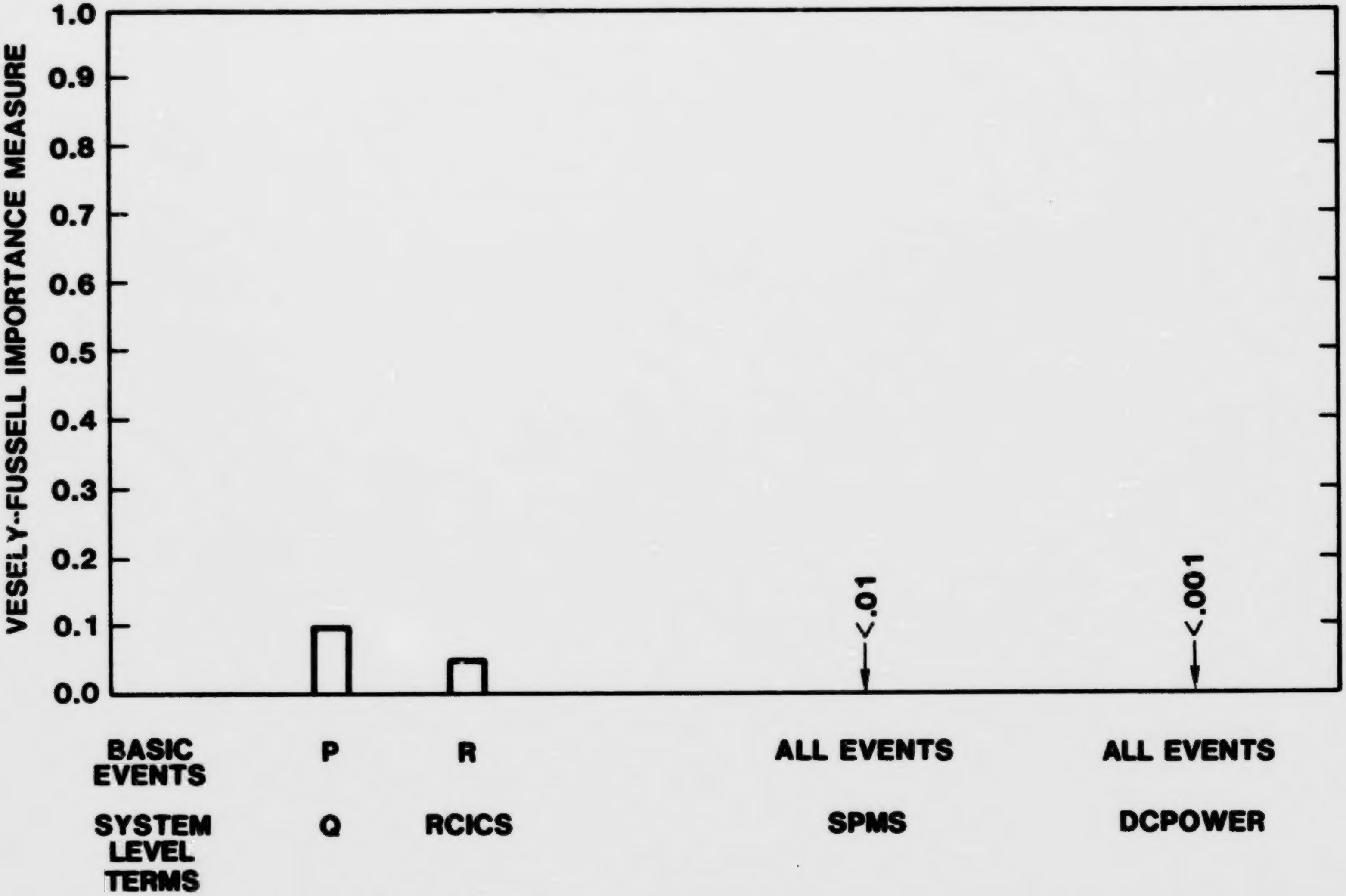

Figure 43. Grand Gulf, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

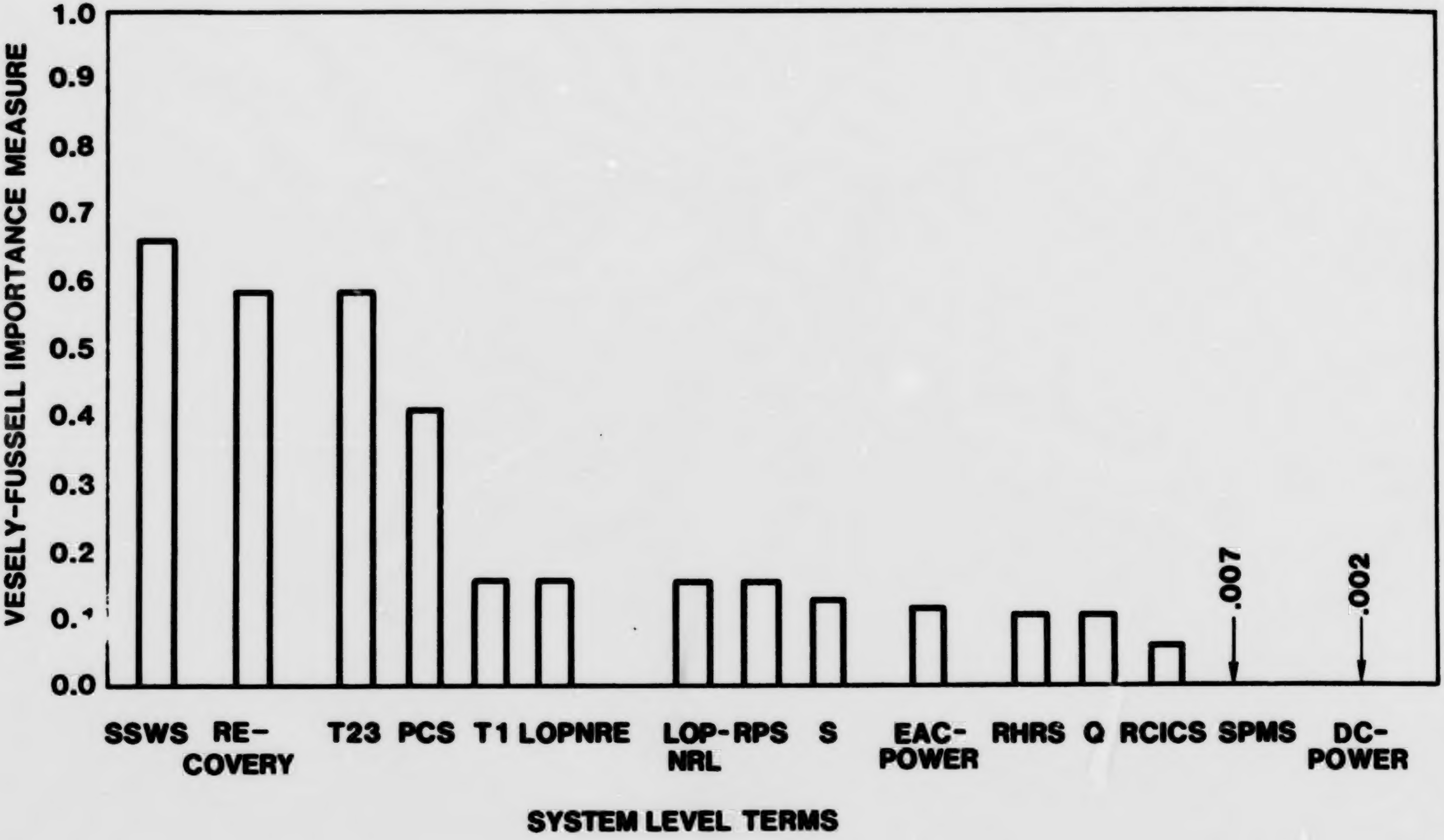

Figure 44. Grand Gulf, Importance of Front Line and Systems to Total Plant Risk: Criterion 3

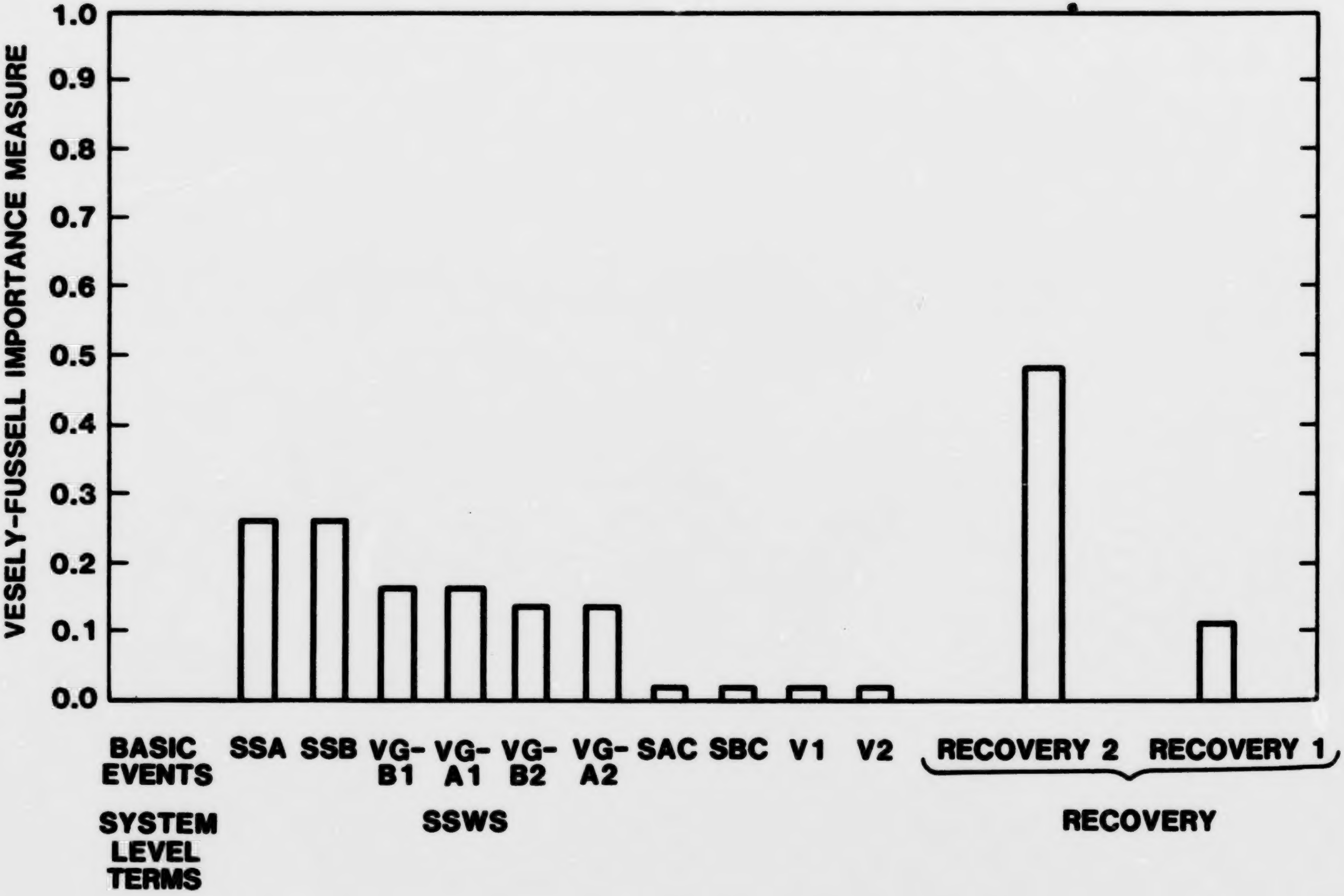

Figure 45. Grand Gulf, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

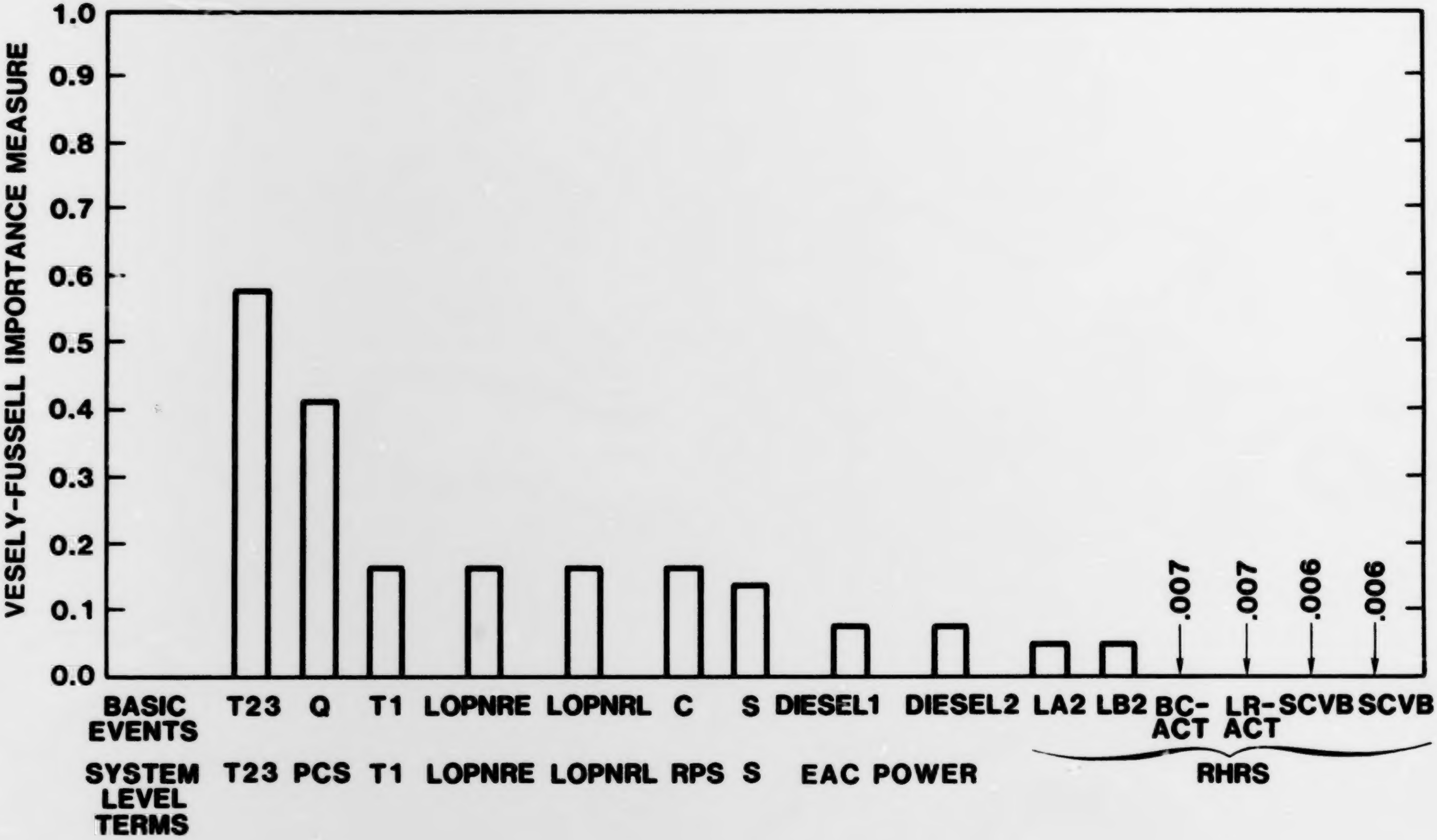

Figure 45. Grand Gulf, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

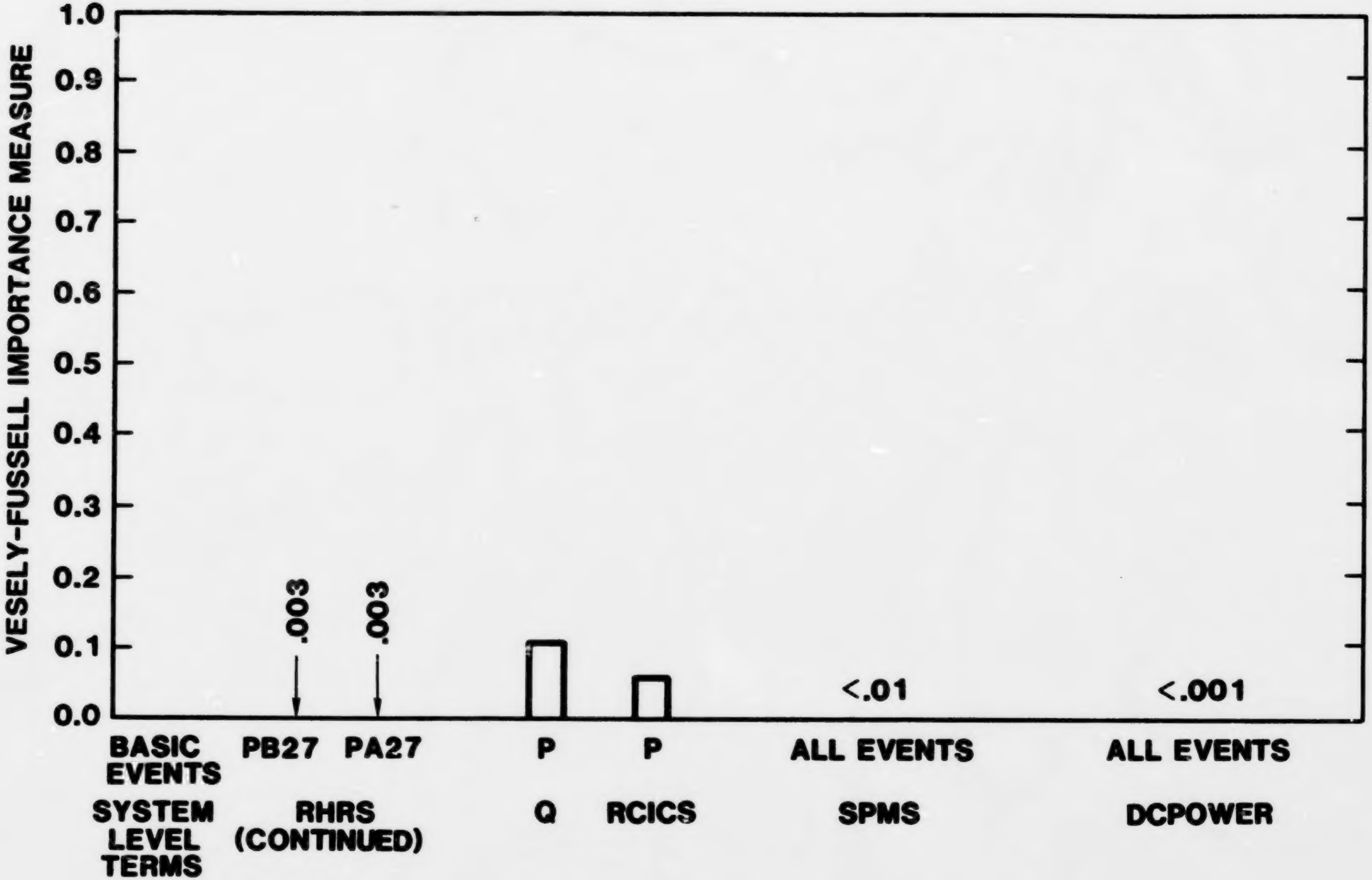

Figure 45. Grand Gulf, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

c. Peach Bottom (WASH-1400 BWR)

The analysis for Peach Bottom was done using WASH-1400.[8] This study was the first PRA performed on a nuclear power plant. The study was done by the NRC. In this study, no sequence fault trees or cut sets were published. For this analysis, simplified fault trees were developed for the dominant sequences by using the sequence event trees to define the system failures of each sequence. The basic events were developed using the descriptions of systems in Appendix II of the PRA. The cut sets for the sequences were developed by solving the simplified fault trees. These sequence fault trees and cut sets are listed in Appendix A12. These cut sets model approximately 87 percent of the core melt frequency and 99 percent of the plant risk.

1. Criteria 1 and 2; Total Core Melt Frequency

The Peach Bottom core melt frequency is totally dominated by two sequences (Table 90). All other sequence frequencies are at least 2 orders of magnitude less than the frequencies of TW and TC.

In modeling transient sequences, WASH-1400 lumped all transients into one term (T). To differentiate between loss of offsite power transient sequences and other transient sequences in this analysis, T was multiplied by RLOP (the fraction of transients with loss of offsite power) in the appropriate cut sets of TW (Appendix A12). To relate the Peach Bottom system level cut sets to RSSMAP initiating events, all cut sets with T and RLOP were labeled as T_1 cut sets at the systems level. All cut sets with just T as an initiating factor were labeled as T_3 cut sets at the system level. The system level results (Tables 93, Figure 46) show transients without loss of offsite power to much more dominant core melt frequency than transients with a loss of offsite power.

The Power Conversion System (PCS) is the most dominant plant system, although the other dominant system terms have significant importances as well. The PCS has two basic events, PCS and PCSLOP. Both involve failure of the PCS several hours after a sequence has been initiated. The unavailability of PCSLOP (Table 92) is greater than that of PCS. However, event PCS is associated with transients not involving loss of offsite power, while PCSLOP is associated with transients with loss of offsite power. The former are much more frequent than the latter, thus PCS has a significantly larger importance than PCSLOP. The PCS is one of three systems which can be used to transfer decay heat to the environment. The other two systems, the RHRS and the High Pressure Service Water System (HPSW), must be used in

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Table 90

Peach Bottom (WASH-1400 BWR)

Plant Vendor	Containment	MWe Rating	PRA Study
General Electric	Mark I	1065	RSS(WASH-1400)

Dominant Accident Sequences

Sequence	Contribution to Overall Criteria Impact Measures		
	1	2	3
1 TW	.56	.56	.61
2 TC	.43	.43	.38

Dominant Accident Sequence Release Category Probabilities

Sequence	Reactor Safety Study BWR Release Categories					Sequence Frequency $(R\text{-}yr)^{-1}$
	1	2	3	4	5	
1 TW	.02	.23	.75			1.7E-5
2 TC	.01		.99			1.3E-5

Total Plant Core Melt Frequency	3.0E-5

Table 91

Peach Bottom System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
HPSW	High Pressure Service Water Failure	HPSW27
LPIS	Low Pressure Injection System	HPCI27
OPERROR	Operator Error	MSD
PCS	Power Conversion System	PCS PCSLOP
RPS	Reactor Protection System	RPS
T_1	Transients With Loss of Offsite Power	T RLOP
T_3	Any Transient Other Than T_1	T

Table 92

Peach Bottom
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
HPSW27	Failure of High Pressure Service Water System After More Than 27 Hours After Initiating Event	1.1E-4
LPCI27	Failure of Low Pressure Coolant Injection System After More Than 27 Hours After Initiating Event	1.2E-4
MSD	Failure of the Operator to Manually Shut Down the Reactor on Failure of the Reactor Protection System	0.1
PCS	Failure of Power Conversion System After More Than 27 Hours After Initiating Event, With Offsite Power Available	7.0E-3
PCSLOP	Failure of Power Conversion System After More Than 27 Hours After Initiating Event With Offsite Power Not Available	2.0E-2
RLOP	Fraction of Transients With Loss of Offsite Power	2.0E-2
RPS	Failure of the Reactor Protection System	1.3E-5
T	Any Transient	10/R-yr

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Table 93

Peach Bottom RSS

Contribution of Front Line and Support Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
T_3	.99	1.00	1.00
PCS	.56	.57	.57
OPERROR	.43	.44	.43
RPS	.43	.44	.43
LPIS	.29	.30	.30
HPSW	.27	.28	.27
T_1	.03	.04	.03

Table 94

Peach Bottom RSS

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Initial Importance	Worst Case Importance	Proportional Importance
T_3	T	.99	1.00	1.00
PCS	PCS	.53	.54	.54
	PCSLOP	.03	.42	.03
OPERROR	MSD	.43	.44	.43
RPS	RPS	.43	.44	.43
LPIS	LPCI27	.29	.30	.30
HPSW	HPSW27	.27	.28	.27
T_1	T	.99	1.00	1.00
	RLOP	.03	.04	.03

Table 95

Peach Bottom RSS

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
T_3	.99	1.00	1.00
PCS	.61	.62	.62
OPERROR	.38	.39	.38
RPS	.38	.39	.38
LPIS	.32	.33	.32
HPSW	.29	.30	.30
T_1	.03	.04	.03

Table 96

Peach Bottom RSS

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Initial Importance	Worst Case Importance	Proportional Importance
T_3	T	.99	1.00	1.00
PCS	PCS	.58	.59	.59
	PCSLOP	.03	.04	.03
OPERROR	MSD	.38	.39	.38
RPS	RPS	.38	.39	.38
LPIS	LPCI27	.32	.33	.32
HPSW	HPSW27	.29	.30	.30
T_1	T	.99	1.00	1.00
	RLOP	.03	.04	.03

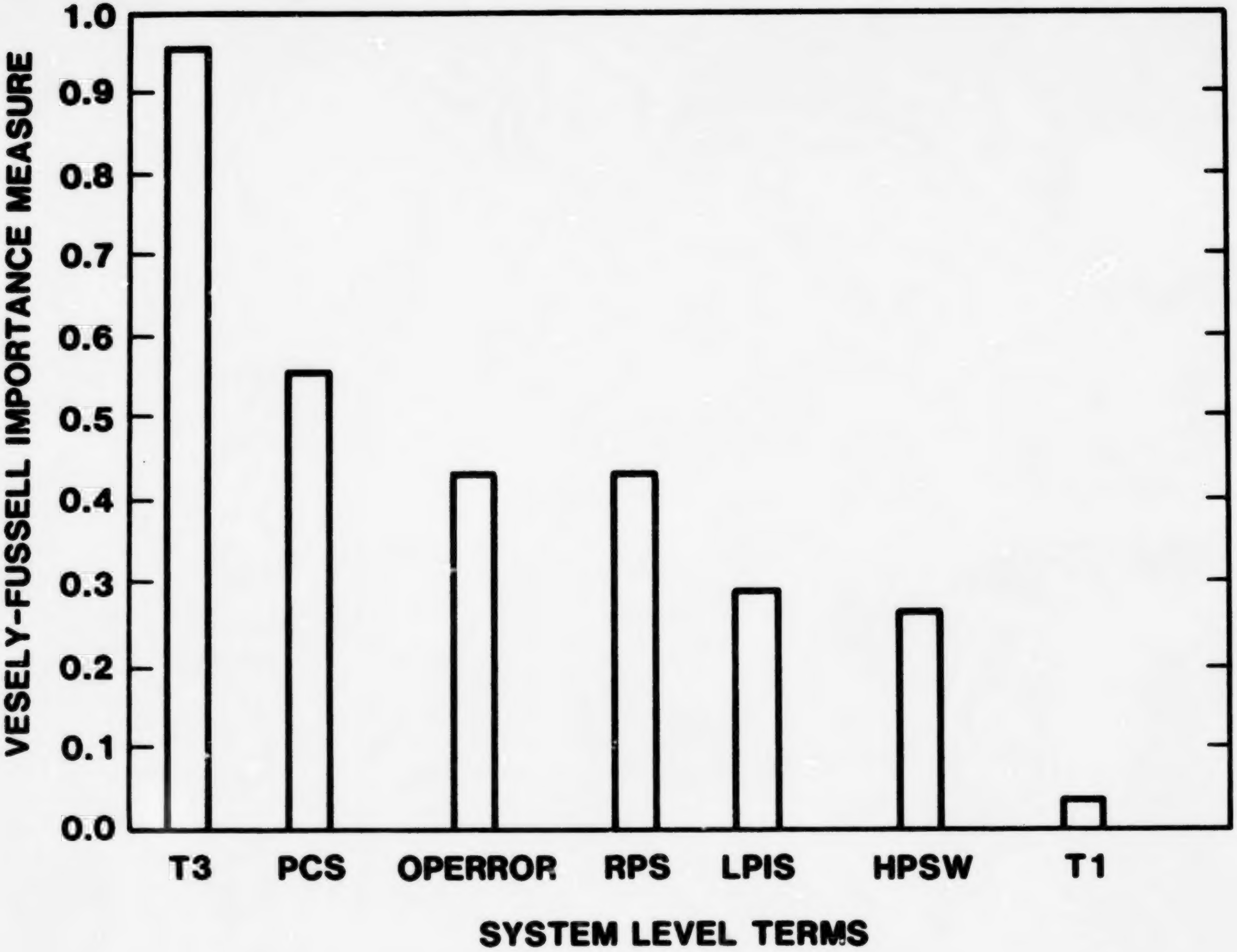

Figure 46. Peach Bottom, Importance of Front Line and Support to Total Melt Frequency: Criteria 1 and 2

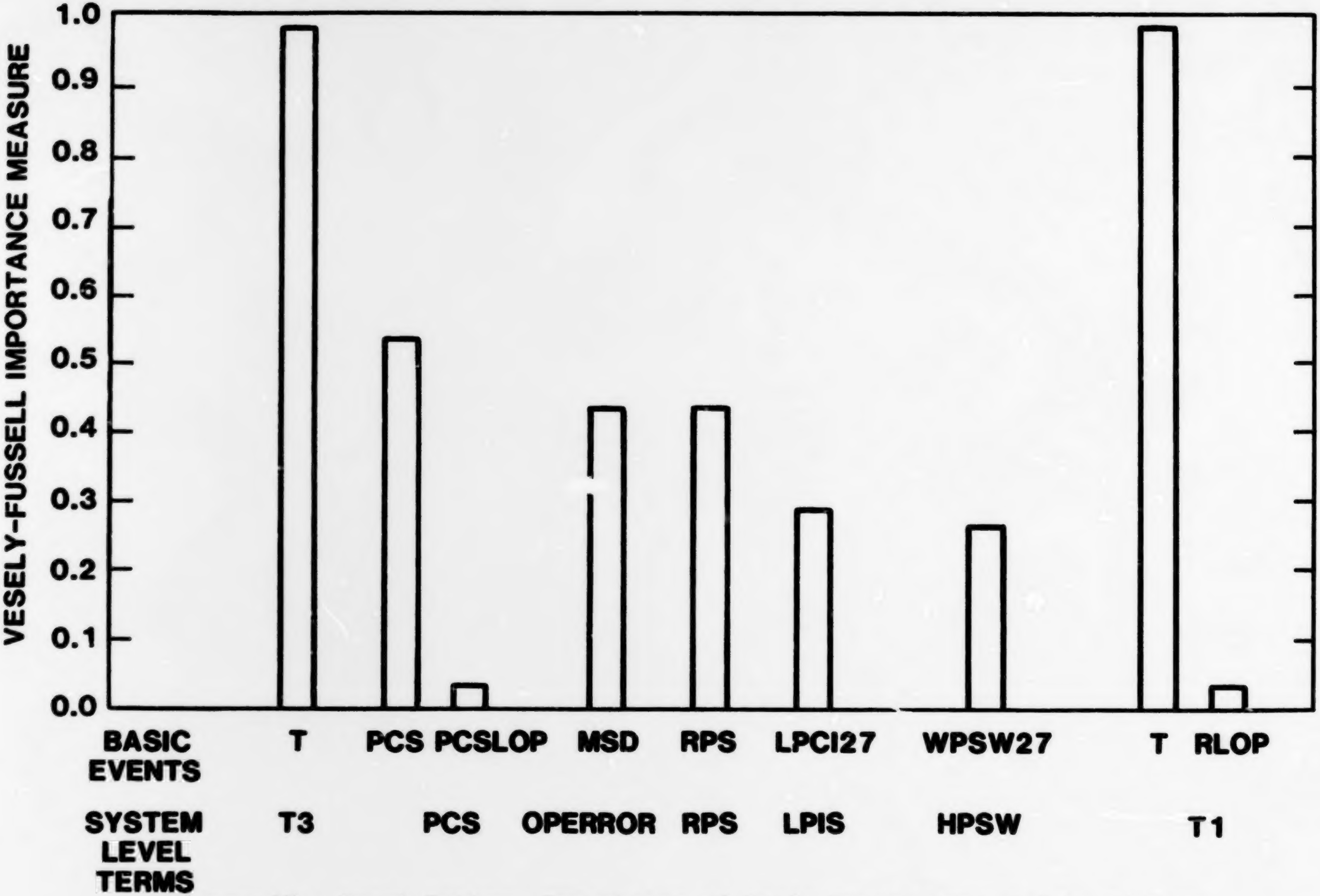

Figure 47. Peach Bottom, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

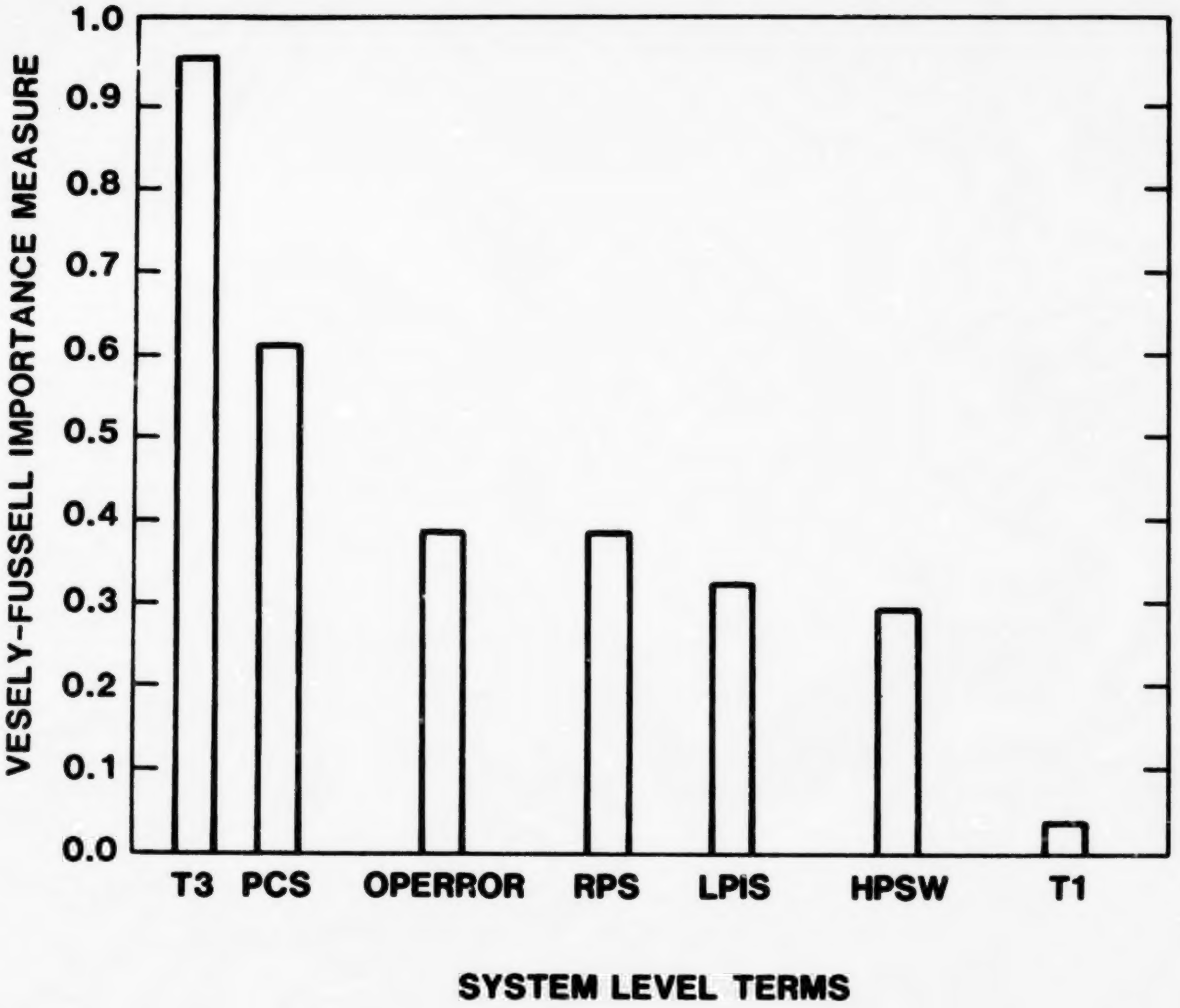

Figure 48. Peach Bottom, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

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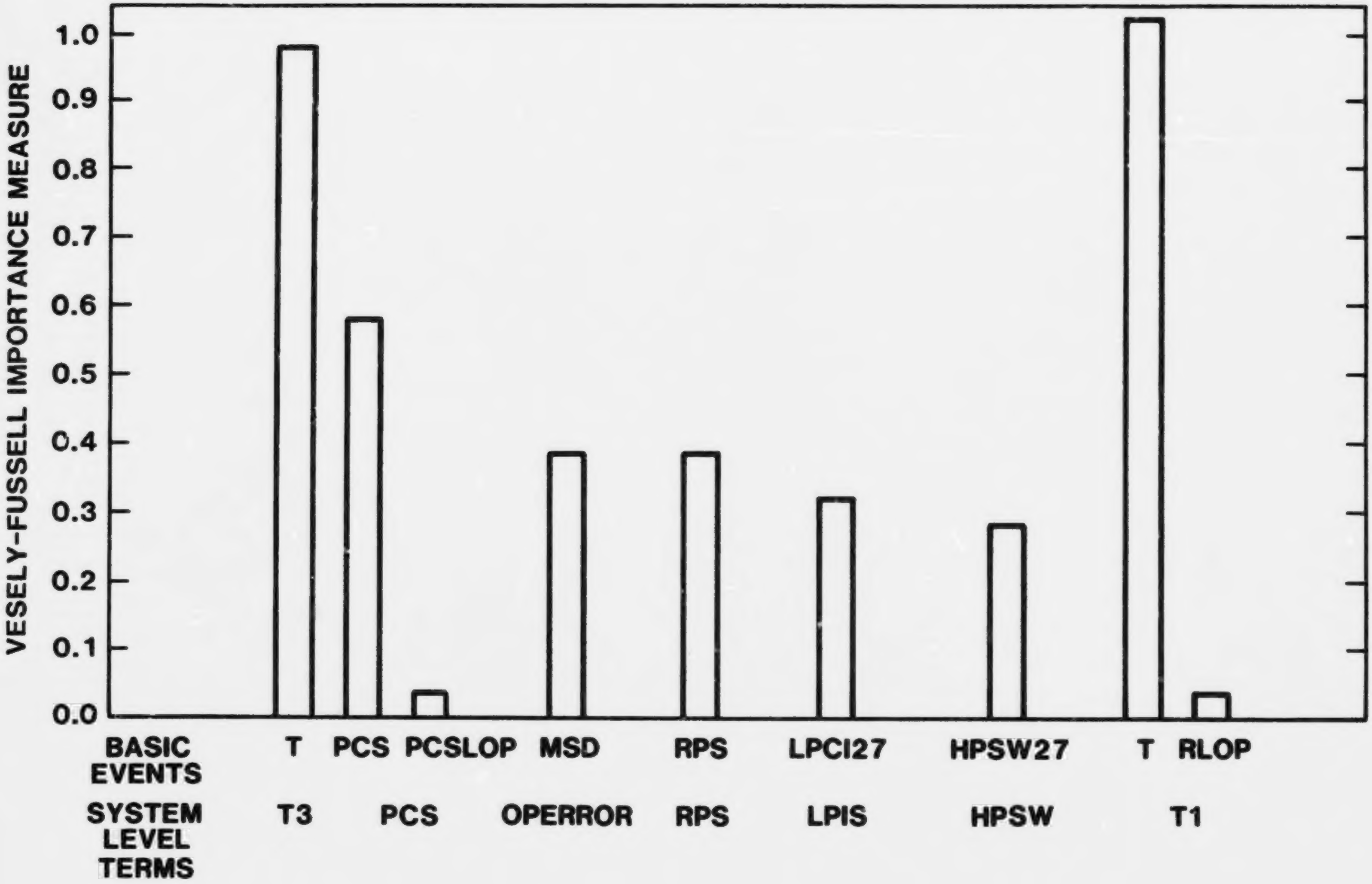

Figure 49. Peach Bottom, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

conjunction with each other. This is event W on the BWR transient event tree, and is described in Appendix V of the PRA under BWR dominant sequences.

Operator errors and failure of the Reactor Protection System (RPS) have equivalent importances, as both have only one basic event each, which occur only in the one cut set sequence TC. The RPS event is called event C in the BWR transient event tree. This event is described in Appendix V of the PRA under BWR dominant sequences.

The LPIS and HPSW failures are system failures which fail the Residual Heat Removal System (RHRS). The LPIS is actually one of the operating modes of the RHRS. The HPSW supplies cooling water to the RHRS heat exchangers. Failure of the HPSW will fail the RHRS. The LPIS and HPSW are described in Appendix II of the PRA.

2. Criterion 3; Total Plant Risk

The results for Criterion 3 are very similar to the results for Criterion 2. The system and basic event importances (Tables 95 and 96, Figures 48 and 49) are only slightly higher than the corresponding values for Criterion two. The same sequences dominate both Criteria, and the release category probabilities for TC and TW are very similar (Table 90). This allows for equal risk weighting of the basic events in both sequences.

d. Limerick

The analysis for Limerick was done using the Limerick Generating Station Probabilistic Risk Assessment.[20] This study was done by Science Applications Incorporated and General Electric for Philadelphia Electric Company. The study did not publish sequence fault trees or cut sets for the dominant sequences. In addition, there was no sequence by sequence description or discussion as in most PRAs. The sequence fault trees and cut sets were therefore developed using the event trees of Section 3.4 of the PRA. The basic events had to be taken off the event trees, as the systems descriptions (Appendix B of the PRA) and the event tree event descriptions (Section 3.4 of the PRA) contain no quantification of events. Only on the event trees can unavailabilities be found. The fault tree model and cut sets from Limerick (Appendix A13) are therefore of an extremely general level. There is virtually a one to one correlation between event tree events, the systems level terms, and the basic events.

There is another significant difference between the Limerick PRA and other PRA's. The frequency of each accident sequence was divided among several containment failure modes specific to the Limerick study. There was no use of the WASH-1400 BWR release categories. However, using information from the PRA and WASH-1400 the Limerick failure modes were correlated to WASH-1400 BWR release categories. This correlation is shown on Table 97. Just as in most other PRAs, there is no difference between Criterion 1 and Criterion 2 of this analysis for Limerick.

1. Criteria 1 and 2; Total Core Melt Frequency

Three sequences are dominant to core melt frequency (Table 98). All three are transient sequences. The cut sets for these sequences model 70 percent of core melt frequency. However, these results should be viewed in light of the fact that the sequence cut sets are at the same level of the event trees. This analysis is very crude and general in nature.

The importance results for both system level terms (Table 101, Figure 50) and the basic events (Table 102, Figure 51) show a virtual one to one correlation. As all the sequences have only one event cut set, and event U (loss of HPIS or Reactor Core Isolation Cooling System (RCICS)) is in each sequence, its dominance to core melt frequency is not suprising. The Feedwater System (FWS) faults, K and Q, are both event Q on the event trees of the PRA. Here, they are differentiated between failure to recover FWS after transient T_P (K)and unavailability of FWS upon demand after other transients (Q).

All the systems (and basic events) involved in the dominant sequences have significant contributions to core melt frequency. However, no further insight into the system importances can be derived. No quantification or description of system unavailabilities or failure are given in the PRA.

2. Criterion 3; Total Plant Risk

There is virtually no difference between results from Criterion 3 and Criterion 2. The same sequences dominate both criteria, and the release category probabilities for each sequence are the same. Thus weighting the cut sets by risk will not cause any change in the ranking of systems and basic event importances to plant risk. The results to plant risk are listed on Tables 103 and 104, and are shown on Figures 52 and 53.

Table 97

Correlation of LGS Containment Failure Modes to WASH-1400 Release Categories

LGS Containment Failure Mode	WASH-1400 Release Cateory
α -	1 from Table 3.5.13 LGS PRA
β -	2 from Table 3.5.13
μ' -	2 from Table 3.5.13
μ -	2 In LGS PRA, μ is combined with γ, Table 3.5.3-3.5.6. γ is correlated to WASH-1400 mode γ' which falls in WASH-1400 release category 2, by comparing description of γ in LGS PRA with γ in WASH-1400.
δ -	4 Correlation of radionuclide releases from table D.2 of LGS PRA with releases from WASH-1400 Table 5.1, plus correlation of description of δ in both PRA's.
γ -	2 see μ
γ' -	3 from comparing descriptions of γ' in LGS PRA and description of γ in WASH-1400.
γ" -	2 Similar to γ', but with less "filtering" from suppression pool, these consequences should be greater than γ'.
ζ -	4 Correlation of release fractions of radionuclides in LGS PRA Tables D.3, D.2 to Table 5.1.
ζc -	4 Correlation of Table D.3 LGS PRA to Table 5.1 of WASH-1400.
δc -	4 Correlation of Table D.3 LGS PRA to Table 5.1 of WASH01400.

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Table 98

Limerick

Plant Vendor	Containment	MWe Rating	PRA Study
General Electric	Mark II	1055	Independent

Dominant Accident Sequences

Sequence	Contribution to Overall Criteria Impact Measures 1	2	3
1 T_TQUX	.05	.05	.04
2 T_FQUX	.25	.25	.25
3 T_EUV	.40	.40	.41

Dominant Accident Sequence Release Category Probabilities

Sequence	Reactor Safety Study BWR Release Categories 1	2	3	4	5	Sequence Frequency $(R\text{-}yr)^{-1}$
1 T_TQUX	.001	0.3	0.3	0.4		7.7E-7
2 T_FQUX	.001	0.3	0.3	0.4		3.6E-6
3 T_EUV	.001	0.3	0.3	0.4		5.9E-6

Total Plant Core Melt Frequency 1.5E-5

Table 99

Limerick Generating Station System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
ADS	Automatic Depressurization System	X XX
FWS	Feedwater System	K Q
HPIS	High Pressure Injection System	U
LPIS	Low Pressure Injection System	VV
T_1	Loss of Offsite Power Transient	TE
T_2	Main Steam Isolation Valve and Loss of Feedwater	TF
T_3	Inadvertent Opening of a Relief Valve/ Turbine Trip Transient	TI TT

Table 100

Limerick
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
K	Loss of Feedwater Not Unrecovered Given Loss of Feedwater Transient	.21/R-yr
Q	Feedwater System Unavailable	2.0E-2
TE	Loss of Power Transient	5.3E-2/R-yr
TI	IORV Transient	.07/R-yr
TF	Main Steam Isolation Valve Closure and Loss of Feedwater	1.78
TT	Turbine Trip Transient	3.98/R-yr
U	High Pressure Coolant Injection or Reactor Core Isolation Cooling System Unavailable	4.9E-3
V	Low Pressure Emergency Core Cooling (LPECCS) System Unavailable	7.7E-5
VV	LPECCS Available	2.3E-2
X	Timely Automatic Depressurization System (ADS) Actuation	2.0E-3
XX	Timely ADS Actuation. Slightly Higher Than X, Due to Possible Reluctance of Operator to Depend on Diesels to Power Low Pressure Pumps, or Inability of Diesels to Start and Run Full Load	2.7E-3

Table 101

Limerick Generating Station

Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
HPIS	.70	1.0	1.0
T_1	.40	.70	.57
LPIS	.40	.70	.57
FWS	.30	.60	.43
ADS	.30	.60	.43
T_2	.25	.55	.35
T_3	.05	.35	.07

Table 102

Limerick Generating Station

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
HPIS	U	.70	1.0	1.0
T_1	TE	.40	.70	.57
LPIS	VV	.40	.70	.57
FWS	K	.25	.55	.35
	Q	.05	.35	.07
ADS	X	.30	.60	.43
T_2	TF	.25	.55	.35
T_3	TT	.05	.35	.07

Table 103

Limerick Generating Station

Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
HPIS	.72	1.0	1.0
T_1	.41	.69	.57
LPIS	.41	.69	.57
FWS	.31	.59	.43
ADS	.31	.59	.43
T_2	.25	.53	.35
T_3	.05	.33	.07

Table 104

Limerick Generating Station

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
HPIS	U	.72	1.0	1.0
T_1	TE	.41	.69	.57
LPIS	VV	.41	.69	.57
FWS	K	.25	.53	.35
	Q	.05	.33	.75
ADS	X	.31	.59	.43
T_2	TF	.25	.53	.35
T_3	TT	.05	.33	.07

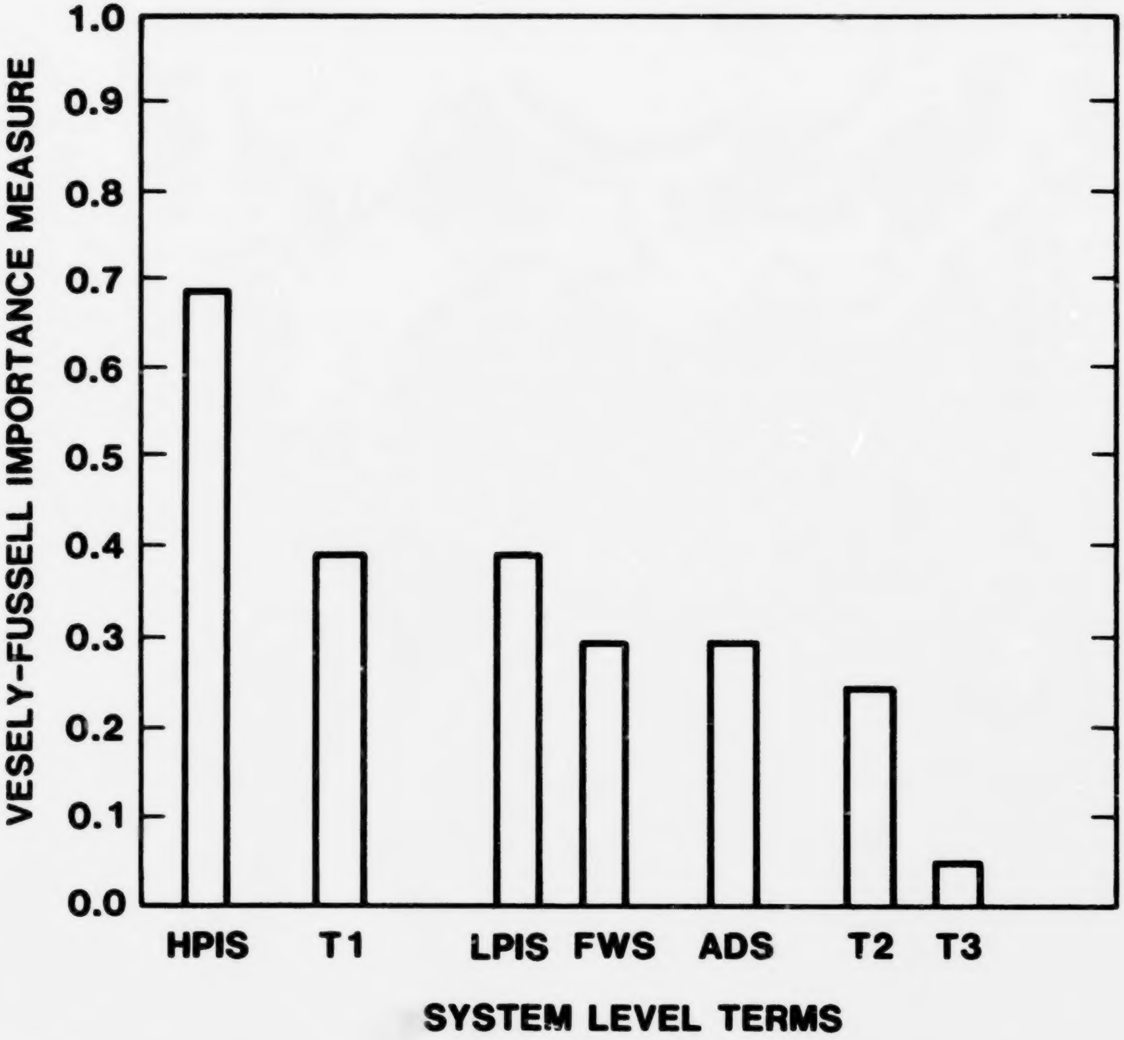

Figure 50. Limerick, Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

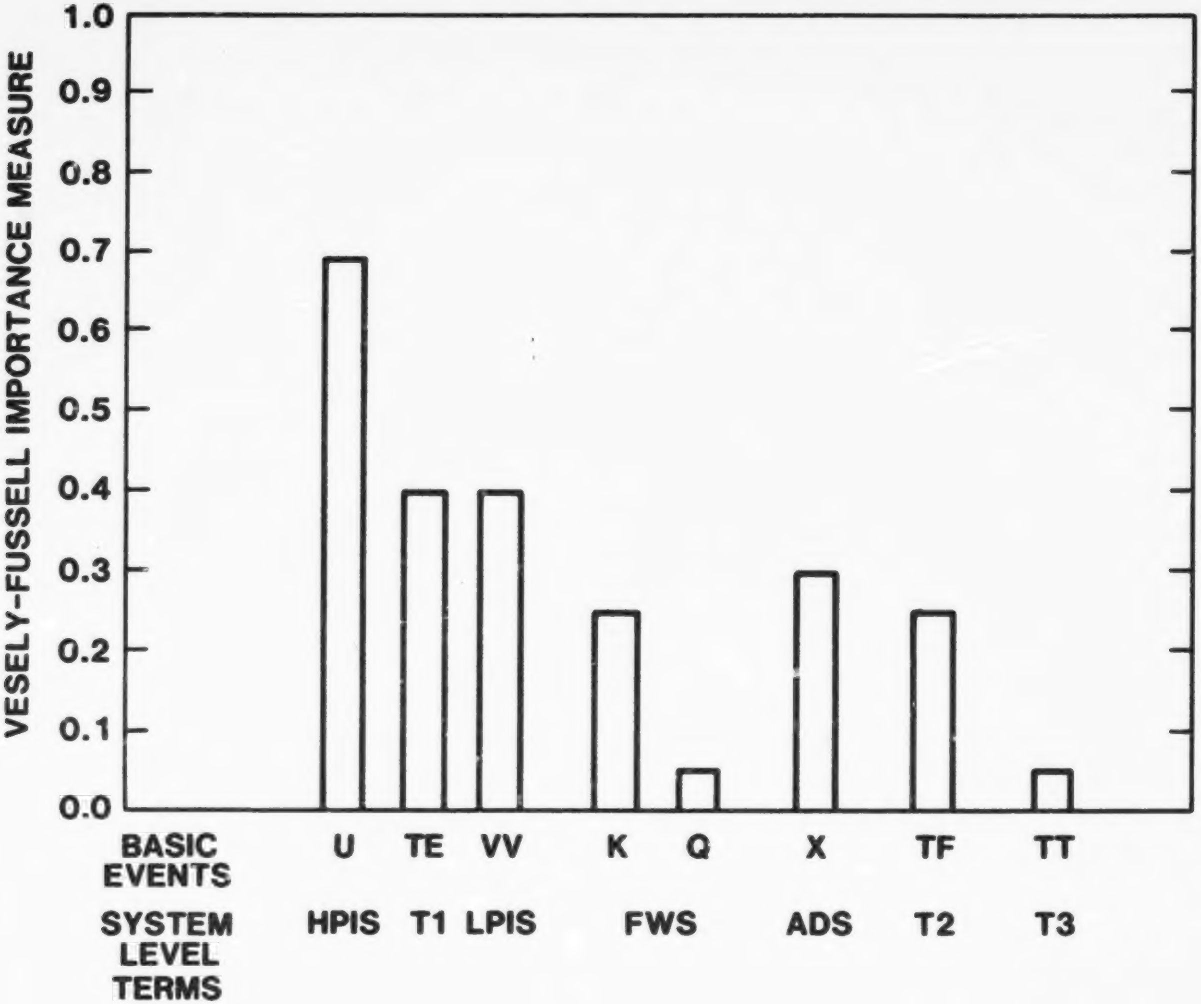

Figure 51. Limerick, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

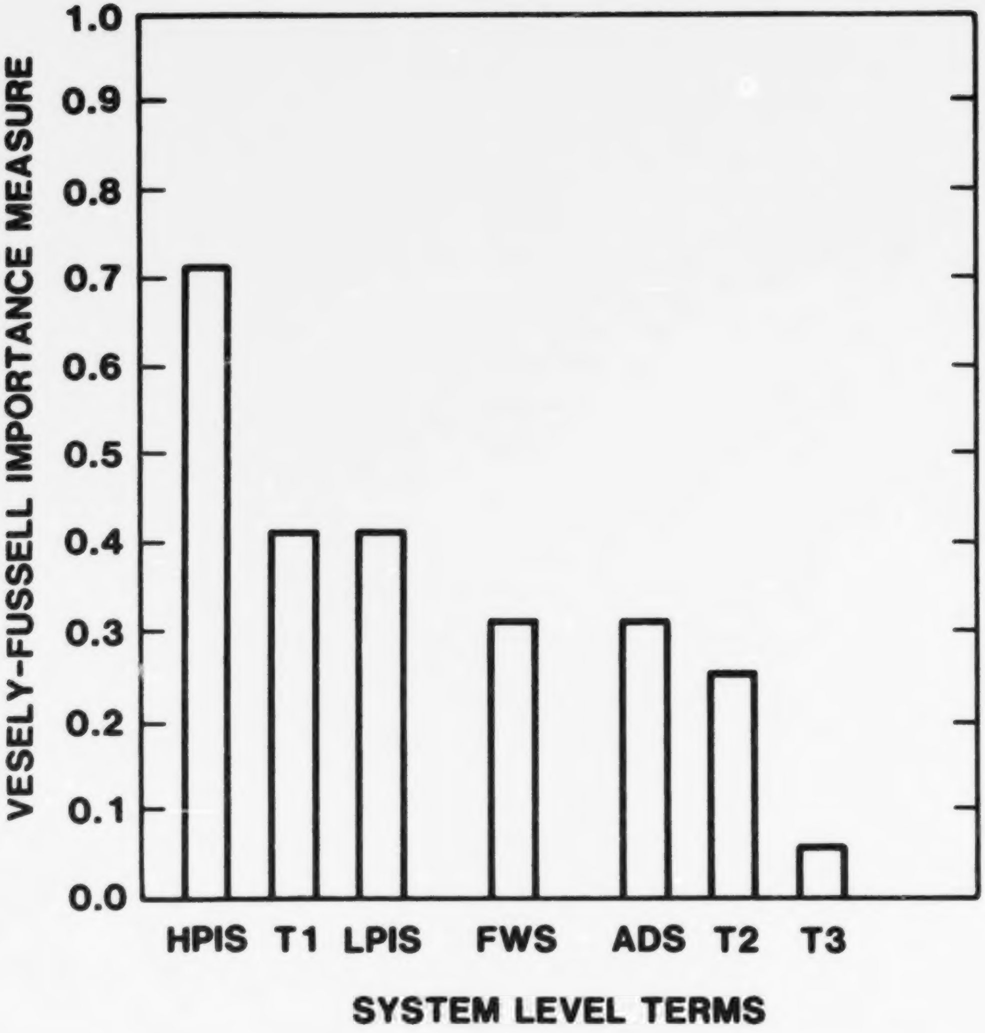

Figure 52. Limerick, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

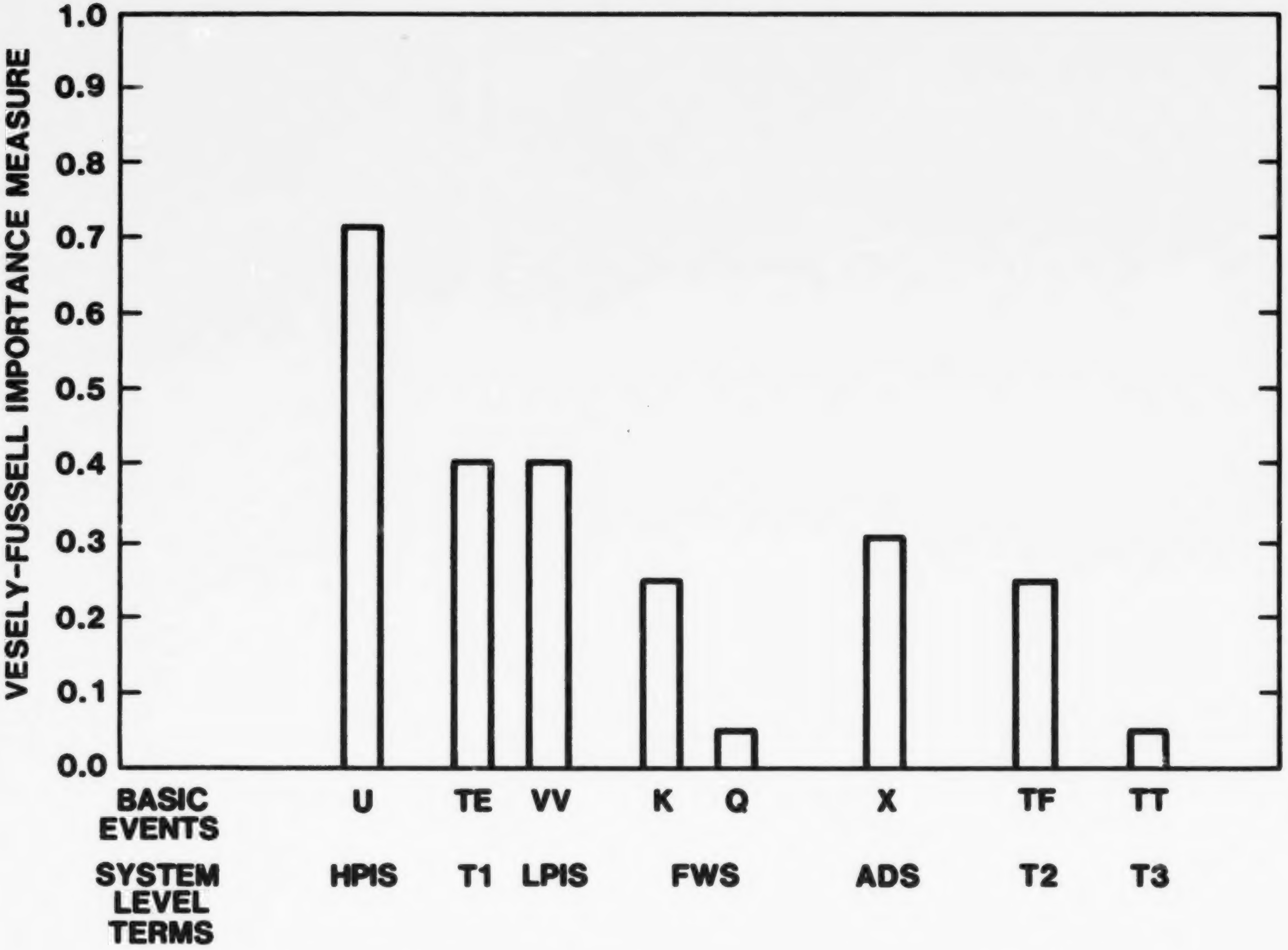

Figure 53. Limerick, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

e. Big Rock Point

The analysis for Big Rock Point was done using the Consumers Power Company Probabilistic Risk Assessment for Big Rock Point Plant.[21] Sequence fault trees and cut sets were not published in the PRA. Simplified fault trees and cut sets were developed using the descriptions of dominant sequences on Table I.2 of Appendix I of the PRA. The basic events were also taken from the sequence descriptions on Table I.2, as no detailed quantification of system failures is provided in the PRA. The cut sets derived for the dominant sequences model approximately 64 percent of total core melt frequency (Criteria 1 and 2) and 60 percent of plant risk. it must be emphasized that the level of cut set development is very general and crude in nature. The cut sets are essentially at the event tree level. The fault trees and cut sets are listed in Appendix A14.

The Big Rock Point PRA does not use WASH-1400 release categories, but uses its own release categories. This presents no problem for Criterion 1 analysis, since the PRA assumes a core melt sequence will fail the containment. Criterion 2 analysis is also unaffected. However, the BWR release category risk factors used for all the other PRAs were developed for WASH-1400 categories, and are useless for the Big Rock Point PRA. Big Rock Point specific risk factors were developed, in the same manner as were the WASH-1400 release category risk factors for this report, using release category consequence data on Table 6.4 of the main report of the PRA.

1. Criteria 1 and 2; Total Core Melt Frequency

Five sequences dominate core melt frequency (Table 97). These sequences are initiated by a steam line break (S_4C), interfacing system LOCA (I_LEML), fire (DPA), a loss of offsite power transient (PEMFSKC) and loss of instrument air (UEMUKC). The system level importance results (Table 100, Figure 54) are very similar to the basic event importances (Table 101, Figure 51). This reflects the general nature of the cut sets developed for Big Rock Point. Only operator errors and fire events have more than one basic event.

The most dominant basic event, KEM, is a failure of a safety relief valve to reseat. This is an event tree level event in the various event trees of the PRA. This event is labeled K in the event trees, but is given the system level lable of Q to correlate it to stuck open relief valve events in other PRAs.

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The next three dominant basic events are the two fire events (DP and A) and an operator error. The DP event is the occurrence of a fire in the cable penetration area inside the containment. Event A (labeled α in the PRA) is failure to suppress the fire manually. These two events comprise a two-event sequence. Fire sequences are discussed in Appendix VI of the PRA.

The operator errors are OPVEC-1, failure to send a man into the containment to open valve VEC-1, and DMWP2B, failure of operator to switch the demineralized water pump over to emergency ac power after loss of offsite power or instrument air, and are discussed in Appendix III. Valve VEC-1 is part of the fire protection system which provides fire water to the emergency condenser. However, here the fire system is used to supply makeup water to the emergency condenser.

If a man is sent into the containment, there is still a probability the valve VEC-1 fails to open. This is illustrated by the importance of event VEC-1.

The remaining events with significant importance, U-NREM, FSEM, IL, and PIS are not discussed or quantified in the PRA. These events are derived from Table I.2 of Appendix II of the PRA.

2. Criterion 3; Total Plant Risk

The results for Criterion 3 are listed on Tables 102 and 103 and shown on Figures 56 and 57. Only four sequences dominate plant risk (Table 97).

Most events are slightly less important to risk than to core melt frequency. This is due to the large fraction of release category probabilities in low risk release categories (Table 97). The exceptions are the fire events. Due to the large release category probability of sequence DPA in release category 3, the fire terms are by far the most dominant events to risk.

Table 105

Big Rock Point

Plant Vendor	Containment	MWe Rating	PRA Study
General Electric	Pre-Mark	75	Independent

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures		
		1	2	3
1	S_4C	.10	.10	.06
2	UE_MUKC	.12	.12	.07
3	I_LEML	.09	.09	-
4	D_PA	.19	.19	.48
5	PE_MF_SKC	.10	.10	.05

Dominant Accident Sequence Release Category Probabilities

Sequence		Big Rock Point BWR Release Categories					Sequence Frequency
		1	2	3	4	5	$(R\text{-}yr)^{-1}$
1	S_4C			.23	.01	.76	1.0E-4
2	I_LEML			.20	.01	.79	8.3E-5
3	D_PA			1.00	.01		1.8E-4
4	PE_MF_SKC			0.20	.01	.79	9.7E-5
5	UE_MUKC			0.20	.01	.79	1.2E-4

Total Plant Core Melt Frequency 9.8E-4

Table 106

Big Rock Point System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
FIRE	Fire Induced Events	A DP
FIRESYS	Fire Suppression System	VEC-1
LOPNRE	Failure to Restore Offsite Power Early	FSEM
OPERROR	Operator Error	DMWP2B OBVPV OPMSIZ OPVEC-1
PCS	Power Conversion System	BC
PIS	Post Incident System	PIS
Q	Safety Relief Valve Failure to Reseat	KEM
RECOVERY	Failure to Restore Instrument Air After Emergency Condenser Failure	U-NREM
S_1	Medium Steamline Break Inside Containment, RSSMAP S_1 LOCA	S_4
T_1	Loss of Offsite Power	LOSP
T_3	Spurious Turbine Bypass Valve Opening	B
U	Loss of Instrument Air	U
V	Interfacing System LOCA	IL

Table 107
Big Rock Point
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
A	Fire Not Suppressed by Manual Action	.10
ACLOCA	Fraction of Interfacing LOCAs Above Core	
B	Spurious Turbine Bypass Valve Opening	.10
BCLOCA	Fraction of Interfacing LOCAs Below Core	
DMWP2B	Failure to Transfer Demineralized Water Pump and Air Compressor to the 2B Bus Given Failure of Emergency Condenser	.25
DP	Fire in Cable Penetration Area (inside containment) Which Affects All Safety System Cables	1.8E-3
FSEM	Failure to Restore Offsite Power and Operate Main Condenser Prior to Safety Valve Lifting	.15
IL	Interfacing System LOCA Due to Failure of Single Valve Isolation Line in Receiver System and Shut Down Cooling	2.0E-3
KEM	Safety Valve Sticks Open Given Emergency Condenser Fails by Valves Failing to Open.	6.2E-2
LOSP	Loss of Offsite Power Due to Line Problems or Auxiliary Bus Problems.	.13
OPBPV	Failure of Operator to Close Bypass Isolation Valve in Time to Prevent Reactor Depressurization System (RDS) (1-2 minutes)	7.7E-1
OPMSIZ	Failure of Operator to Manually Close Main Steam Isolation Valve	.13
OPVEC-1	Failure to Send Operator Into Containment to Open VEC-1	.23

Table 107 (Cont.)

Big Rock Point
Boolean Terms From Sequence Fault Trees

Boolean Term	Description	Value
PIS	Failure of Post Incident System	1.E-2
S4	Medium Steam Line Break Inside Containment	1E-4
U	Loss of Instrument Air	.06
U-NREM	Failure to Restore Instrument Air Given Emergency Condenser Fails Due to Lack of Makeup Water Reaching it	.10
VEC-1	Inability of Operator to Open VEC-1	.077
PIS10M	Failure of Post Incident System to Succeed Throughout a 10-Month Mission Required in the Event of an Interfacing LOCA Below the Core	4.5E-2

Table 108

Big Rock Point

Contribution of Front Line and Support Systems To Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
Q	.21	.63	.36
FIRE	.18	.60	.32
OPERROR	.18	.60	.31
U	.12	.53	.20
RECOVERY	.12	.53	.20
S_1	.10	.52	.18
T_1	.09	.51	.16
LOPNRE	.09	.51	.16
V	.08	.50	.15
FIRESYS	.05	.47	.09
PIS	.002	.42	.004

Table 109

Big Rock Point

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Core Melt Frequency: Criteria 1 and 2 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
Q KEM	.21	.63	.36	
FIRE	DP	.18	.60	.32
	A	.18	.60	.32
OPERROR	OPVEC-1	.18	.58	.27
	DMWP2B	.09	.51	.16
U	U	.12	.53	.20
RECOVERY	U-NREM	.12	.53	.20
S_1	S_4	.10	.52	.18
T_1	LOSP	.09	.51	.16
LOPNRE	FSEM	.09	.51	.16
V	IL	.08	.50	.15
FIRESYS	VEC-1	.05	.47	.09
PIS	PIS	.002	.42	.004

Table 110

Big Rock Point

Contribution of Front Line and Support Systems To Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Importance	Worst Case Importance	Proportional Importance
FIRE	.48	.82	.73
Q	.11	.45	.17
OPERROR	.10	.44	.15
U	.06	.40	.09
RECOVERY	.06	.40	.09
S_1	.06	.40	.09
T_1	.05	.39	.08
LOPNRE	.05	.39	.08
FIRESYS	.03	.37	.04

Table 111

Big Rock Point

Contribution of Basic Events, Grouped by Front Line and Support System, to Total Plant Risk: Criterion 3 (Vesely-Fussell Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
FIRE	DP	.48	.82	.73
	A	.48	.82	.73
Q	KEM	.11	.45	.17
OPERROR	OPVEC-1	.08	.43	.13
	DMWP2B	.05	.39	.08
U	U	.06	.40	.09
RECOVERY	U-NREM	.06	.40	.09
S_1	S_4	.06	.40	.09
T_1	LOSP	.05	.39	.08
LOPNRE	FSEM	.05	.39	.08
FIRESYS	VEC-1	.03	.37	.04

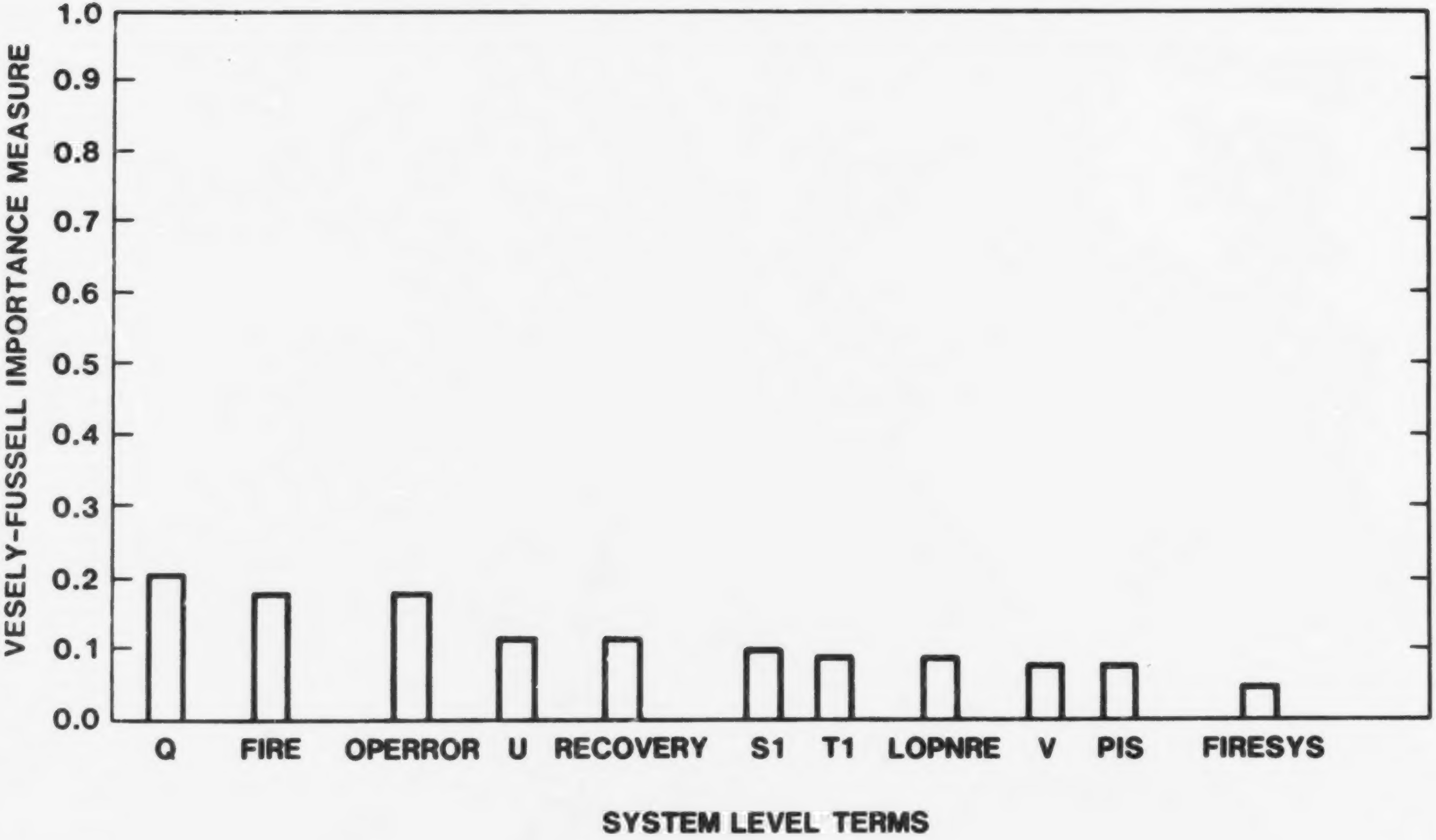

Figure 54. Big Rock Point, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

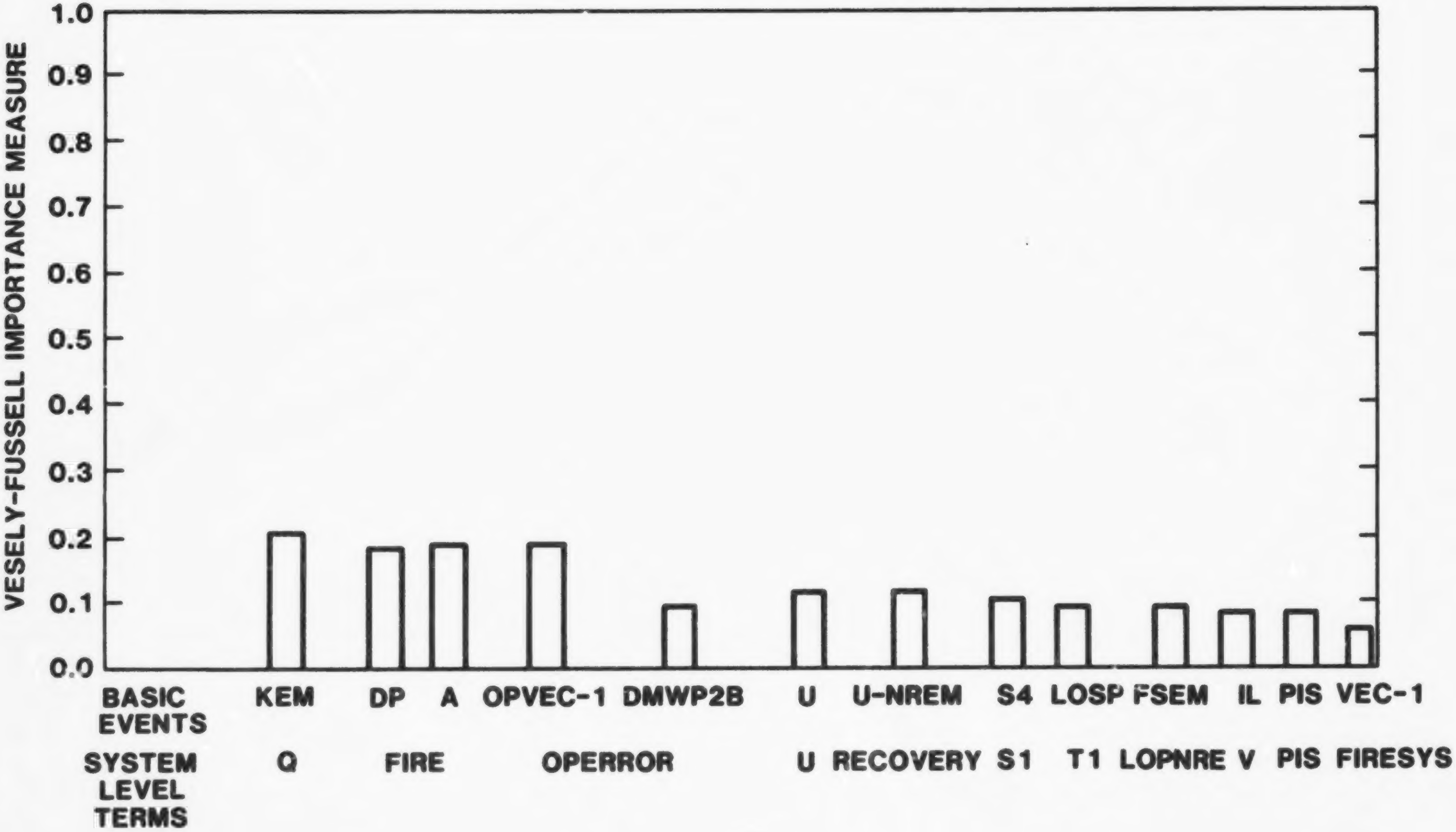

Figure 55. Big Rock Point, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

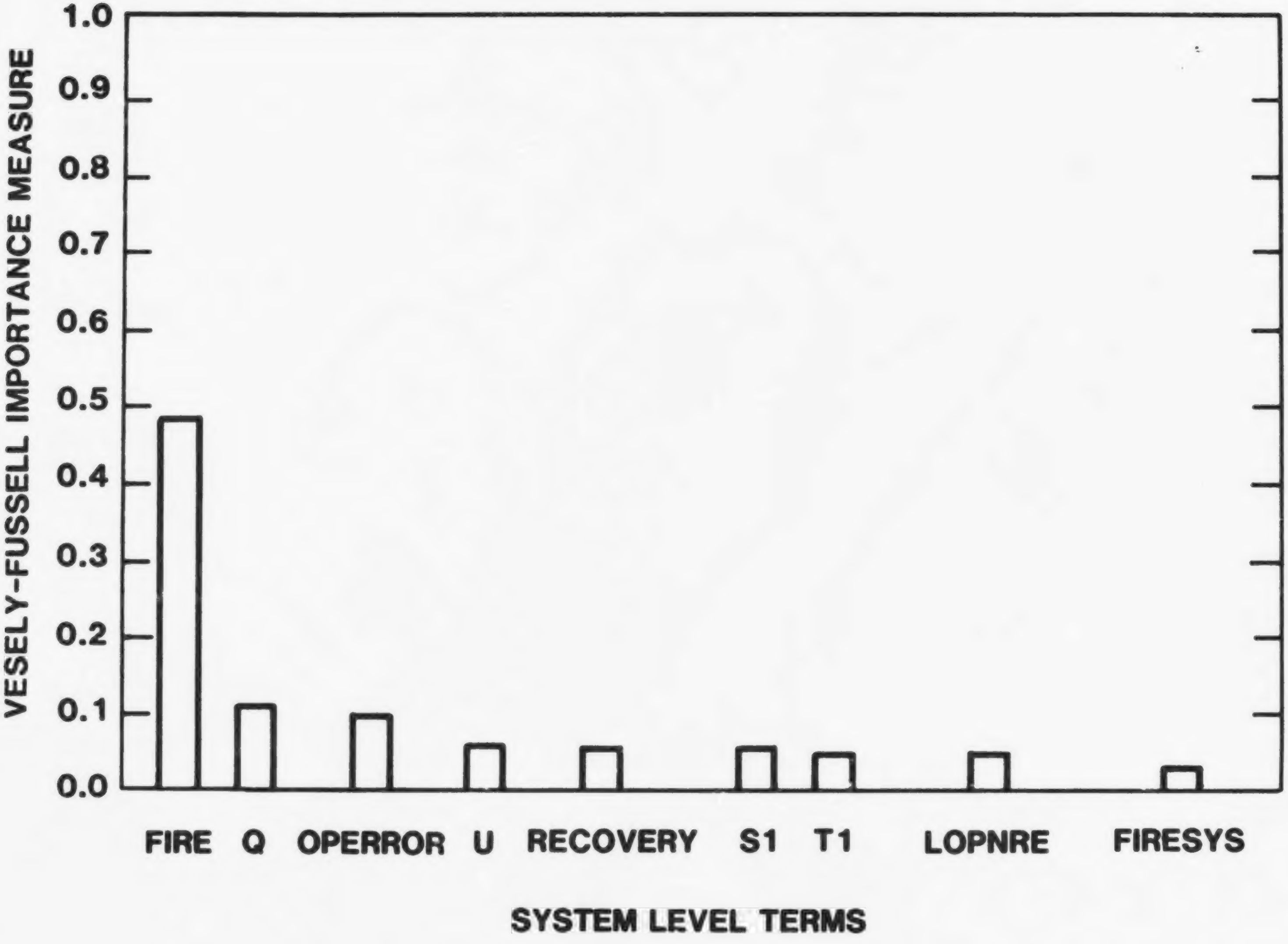

Figure 56. Big Rock Point, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

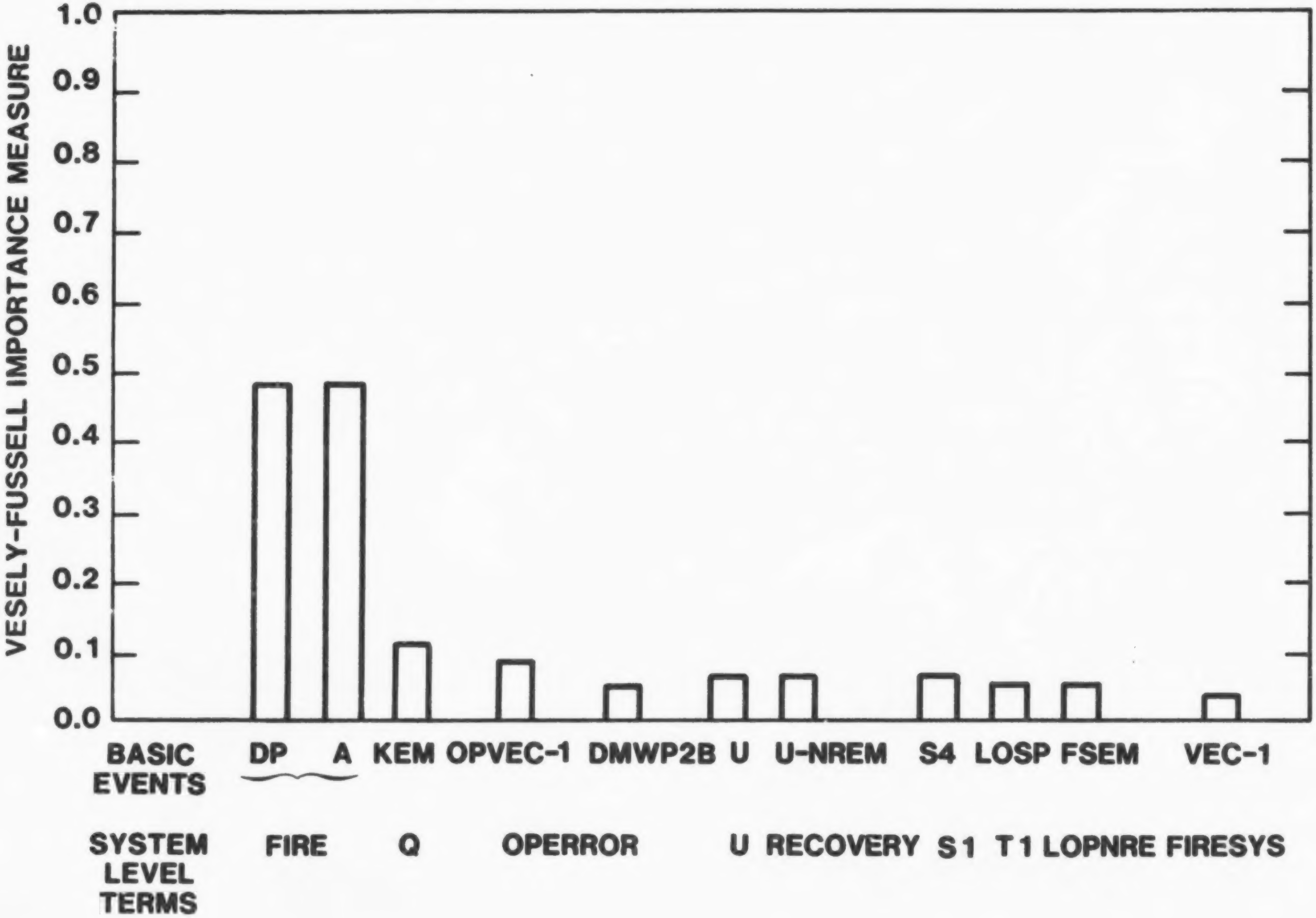

Figure 57. Big Rock Point, Importance of Events, Grouped by Systems, to Total Plant Risk: Criterion 3

f. Millstone 1

The analysis for the Millstone 1 plant was done using the Millstone Interim Reliability Evaluation Program (IREP) PRA.[23] Cut sets for the dominant sequences were developed by expanding the "super" Boolean terms of the dominant sequence cut sets published in the PRA. These "super" terms are defined in terms of basic events for each sequence. It is these basic events which are used in the expanded cut sets. The basic events are listed and described in Table 114. The Millstone PRA's used a labeling system for basic events which coded a basic event's label by its system, component type, component number, and failure mode. It is for this reason that the Millstone basic event names are so cumbersome. The labeling system of the PRA is described in Appendix C of the PRA. The cut sets analyzed for this analysis model are 50 percent of core melt frequency and 52 percent of plant risk. The dominant cut sets for each sequence are listed in Appendix A15. The source of the modeling gap for Millstone is twofold. By analyzing only those sequences contributing 5 percent or more to total core melt frequency or plant risk, approximately 80 percent of core melt frequency and plant risk were considered. The remaining gap is due to the limited number of cut sets published in the PRA.

1. Criteria 1 and 2; Total Core Melt Frequency

Table 112 shows that seven sequences dominate total core melt frequency. Six of the seven are initiated by loss of offsite power (T_4) followed by failure to keep the core cooled at high pressures. The other dominant sequence involves a loss of power conversion system (PCS) initiation followed by failure of reactor to scram.

The system level importance results (Table 115, Figure 58) agree very well with the major engineering insights discussed in the PRA summary. The major system level contributions are loss of offsite power transient (T_4), failure to recover offsite power within 1/2 hour (.5HR-FTR-OS-PWR), failure of emergency AC systems (EACPOWER), operator failure to manually depressurize the Reactor Cooling System (RCS) (OPERROR), and failure of a safety valve to reset (S/R-VR). Millstone's high pressure emergency cooling systems are highly dependent on the gas turbine emergency power source, which has a relatively low reliability. Furthermore, the plants Automatic Pressure Relief System (APR) is such that it only reduces pressure during a LOCA. Thus, for transient sequences the operator must manually depressurize the RCS to allow the low pressure cooling systems to operate given failure of the high pressure cooling function. The PRA claims that the procedure the operator must follow is confusing, resulting in a high failure rate for this task. Other system dependencies on emergency AC power add to the importance of the loss of power terms. The Low Pressure Coolant Injection System (LPCI) is dependent on both the diesel and gas turbine trains of emergency AC power. The Isolation Condensor Make Up System, part of the Isolation Condensor System (IC), is failed by loss of the gas turbine generator as well. This, in turn, fails the IC.

At the basic event level, Table 116 shows that the single event system level terms T_4, OPERROR, S/R-VR, and .5HR-FTR-OS-PWR are dominant, as expected from the system level results. The emergency AC power system is dominated by a loss of function of the diesel generator, and by several circuit breaker failures which prevent the loading of emergency AC loads onto the gas turbine buses.

2. Criterion 3: Total Plant Risk

Table 112 shows that six sequences are dominant to total plant risk. However, only three sequences which are dominant for Criteria 1 and 2 also are dominant for Criterion 3. Only two of the six sequences, T_4JCD and T_2A, result in melt down at high RCS pressure. The Millstone PRA assigns a much higher probability of containment failure due to in-vessel steams explosions at low pressures (1.0E-2) than at high pressures (1.0E-4), so the dominance of the low pressure melt down sequences to plant risk is not surprising.

The systems level results (Table 117) reflect the dominance of low pressure cooling failures to plant risk. Emergency AC power is very important, due to the high dependence of LPCI on the diesel and gas turbine trains. Failure to recover offsite power is important, however, for low pressure sequences, offsite power must be recovered within 20 hours, as opposed to only 30 minutes or a couple hours for high pressure sequences. Operator failure to manually depressurize the RCS is not very important to risk since only one sequence dominant to risk, T_4JCD, involves this error. Melt down at low pressure implies that the operator succeeded to depressurize the RCS.

Table 112

Millstone

Plant Vendor	Containment	MWe Rating	PRA Study
General Electric	Mark I	652	IREP

Dominant Accident Sequences

Sequence		Contribution to Overall Criteria Impact Measures		
		1	2	3
1	T_4JCD	.23	.23	.08
2	T_4JCEFG	.13	.13	.06
3	T_4KCD	.10	.10	-
4	T_4KCEFG	.10	.10	-
5	T_4LCD	.10	.10	-
6	T_2A	.07	.07	.18
7	T_4JCDG	.07	.07	-
8	T_4JCMG	-	-	.09
9	T_4LCMG	-	-	.09
10	T_4KCMG	-	-	.08

Dominant Accident Sequence Release Category Probabilities

Sequence		Reactor Safety Study BWR Release Categories					Sequence Frequency $(R\text{-}yr)^{-1}$
		1	2	3	4	5	
1	T_4JCD	.0001	0.0	0.1	0.9	0.0	7.0E-5
2	T_4JCEFG	.01	0.0	0.1	1.0	0.0	4.0E-5
3	T_4KCD	.0001	0.0	0.1	1.0	0.0	3.0E-5
4	T_4KCEFG	.01	0.0	0.1	1.0	0.0	3.0E-5
5	T_4LCD	.0001	0.0	0.1	1.0	0.0	3.0E-5
6	T_2A	.0001	0.1	1.0	0.0	0.0	2.0E-5
7	T_4JCDG	.0001	0.0	0.1	1.0	0.0	2.0E-5
8	T_4JCMG	.01	0.1	0.9	0.0	0.0	1.0E-5
9	T_4LCMG	.01	0.1	0.9	0.0	0.0	1.0E-5
10	T_4KCMG	.01	0.1	0.9	0.0	0.0	9.0E-6

Table 113

Millstone
System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
DCPOWER	DC Power System	DC-A-BATT-TMO
		DC-A-BATT-LOF
EACPOWER	Emergency AC Power System	AC-DGN-LOF
		AC-IV-XFR-LOF
		AC-IV-3-FRC
		AC-F5-1F-FRC
		AC-IAC-ABT-LOF
		AC-EF3-ABT-LOF
		AC-1S14C-1-FTC
		AC-1S14A-2-FTO
		AC-14CT-16-FTC
		AC-14AT2-1-FTC
		92662-1-FTE
		92694-1B-FTE
		92694-1A-FTD
		92662-1-1FC
		9262NP-3-BFC
		926LNP-1-3FC
		92GLNP-1-FTE
		92662-1A-XFO
		92662-1-XFO

Table 113 (Cont.)

Millstone
System-Level Grouping of
Boolean Terms From Sequence Fault Trees

System	System Description	Component
FWCI	Feedwater/Feedwater Coolant Injection System	AC-14C-3-FRC
		AC-14C-4-FRC
		AC-14A-3-FRC
		AC-E4-3C-FRC
		AC-14C-15-FRC
		AC-E3-5E-FRC
		MW-97-SOV-FRO
		LCLWS-V2-1A-XFC
		LCLSW-V2-3A-XFC
		LCL94X-3A-FTD
		LCL94X-3B-FTD
		LCL94X-1A-FTD
		LCL27-3-FTD
		LCL94X-1B-FTD
		LCL94X-3C-FTD
		LCL94X-1B-9FO
		LCL94X-3C-1FO
		LCL94X-1A-FTE
		LCLPS-2-8-TOM
		LCLPS-2-16-TOM
		LCLPS-2-39A-TOM
		LCLPS-2-40A-TOM
		LCLPS-2-8-XFC
		LCLPS-2-16-XFC
		LCLPS-2-39A-XFC
		LCLPS-2-40A-XFC
		92694-2A-FTD
		92694-1A-FTD
		92694-1A-1FO
		92694-2A-1FO
		92694-2A-1FO
		92694-1A-5FO
		92694-2A-5FO
		9062-25-50-FTD
		906CS-162-FTD

Table 113 (Cont.)

Millstone
System-Level Grouping of Boolean Terms From Sequence Fault Trees

System	System Description	Component
IC	Isolation Condenser	FTR-ICM-10 ICM-10-MOV-FTO IC-3-MOV-TMC IC-1-MOV-TMC IC-2-MOV-TMC IC-4-MOV-TMC 903595-118A-3FC 903595-118B-3FC
OPERROR	Operator Error	MDP
RPS	Reactor Protection System	A
SWS	Service Water System	AC-EF7-26-FTC SW-448-STR-LOF SW-9-MOV-FTC SW-DGN-HTX-LOF
S/R-VR	Safety/Relief Valve fails to Reset	S/R-VR
T_2	Loss of Power Conversion System Transient	T_2
T_4	Loss of Normal AC Power Transient	T_4
.5HR-FTR-OS-PWR	Failure to Recover Offsite Power Within 5 Hours	.5HR-FTR-OS-PWR
2H-FTR-OS-PWR	Failure to Recover Offsite Power Within 2 Hours	1HR-FTR-OS-PWR
20HR-FTR-OS-PWR	Failure to Recover Offsite Power Within 20 Hours	20HR-FTR-OS-PWR

Table 114

Millstone
Boolean Terms From Sequence Fault Trees

Boolean Terms	Description	Value
AC-15114C-1-FTO AC-1514A-2-FTO AC-14CT-16-FTC AC-14AT2-1-FTC	Circuit breaker failures Prevent Loading of Emergency AC Buses Onto Gas Turbine.	1.7E-2
AC-14C-3-FRC AC-14C-4-FRC AC-14A-3-FRC AC-E4-3C-FRC	Failure of Feedwater/Feedwater Coolant Injection Systems (FWCI) Pump Breaker	1.7E-2
AC-EC-5E-FRC AC-14C-15-FRC	Circuit Breaker Failures in FWCI Fail Make Up Flow to FWCI	1.7E-2
MW-97.SOV-FRO	Solinoid Operated Valve in FWCI Fails Make Up Flow to FWCI	1.7E.2
SW-9-MOV-FTC AC-EF7-26-FTC	Mechanical Failures of Service Water System (SWS) Fail Cooling to FWCI Pumps	1.7E-2
SW-448-STR-LOF	Failure of Strainer in SWS Fails Cooling to FWCI Pumps	2.4E-2
LCLW-V2-1A-XFC LCLW-V2-3A-XFC	Switch Fails to Initiate FWCI	2.3E-3
LCL94X-3A-FTD 92694-2A-5FO 92694-1A-5FO 92694-2A-1FO 92694-1A-1FO LCL94X-1A-FTD LCL94X-3C-1FO 906CS-162-FTD LCL94X-1B-9FO LCL94X-3B-FTD 92694-2A-FTD 92694-1A-FTD LCL94X-1A-FTE LCL94X-1B-FTD LCL94X-3C-FTD LCL27-3-FTD 9062-25-50-FTD	Relay Failures Which Prevent FWCI Initiation	1.7E-3

Table 114 (Cont.)

Millstone
Boolean Terms From Sequence Fault Trees

Boolean Terms	Description	Value
LCLPS-2-8-XFC LCLPS-2-16-XFC LCLPS-2-39A-XFC LCLPS-2-40A-XFC	Instrument Contacts Fail to Initiate FWCI Pumps	2.3E-3
LCLPS-2-8-TOM LCLPS-2-16-TOM LCLPS-2-39A-TOM LCLPS-2-40A-TOM	Pressure Sensor Down for Test or Fail FWCI Pumps Maintenance	1.0E-2
92662-1A-XFO 92662-1-XFO	Loss of Normal Power Trip Signal Relays Fail to Reset, Failing All Emergency AC Power	1.8E-3
IC-3-MOV-TMC	Isolation Condenser (IC) Closed For Test on Maintenance	8.0E-4
IC-1-MOV-TMC IC-2-MOV-TMC IC-4-MOV-TMC	IC Values Closed for Test or Maintenance	2.3E-3
903595-118A-3FC 903595-118B-3FC	IC Initiation Logic Relays Fail	1.8E-3
DC-A-BATT-TMO	DC Battery A Down for Test or Maintenance Failing IC Valves and Diesel Generator	5.0E-4
DC-A-BATT-LOF	Failure of DC Battery A	2.4E-4
92662-1-FTE 92694-1B-FTE 92662-1-1FC 926LNP-3-BFC 92694-1A-FTD 926LNP-1-FTE 926LNP-1-3FC	Relay Failures Prevent Start Signal for Gas Turbine	7.0E-3

Table 114 (Cont.)

Millstone
Boolean Terms From Sequence Fault Trees

Boolean Terms	Description	Value
AC-IV-XFR-LOR	Transformer Failure Fails Instrument AC	.2
AC-IV-3-FRC AC-F5-IF-FRC	Breaker Failures Fail Instrument AC	.1
AC-DGN-LOF	Failure of Diesel Generator	6.0E-2
SW-DGN-HTX-LOF	Failure of SWS Heat Exchanger Fails Diesel Generator	7.14E-2
AC-IAC-ABT-LOF AC-EF3-ABT-LOF	Auto Bus Transfers Fail	.3
MDP	Operation Fails to Manually Depressurize the Reactor Coolant System	7.0E-2
A	Common Mode Mechanical Failures of RPS	9.0E-6
S/R-VR	Safety/Relief Valve Fails to Reset	1.8E-2
T_2	Loss of Power Conversion System Transient	2.14/R-yr
T_4	Loss of Normal AC Power Transient	0.2/R-yr
.5HR-FTR-OS-PWR	Failure to Recover Offsite Power Within .5 Hours	.43
2HR-FTR-OS-PWR	Failure to Recover Offsite Power Within 2 Hours	.24
20HR-FTR-OS-PWR	Failure to Recover Offsite Power Within 20 Hours	.05

Table 115

Millstone
Contribution of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2
(Vesely-Fussell Importance Measure)

System, Operator Action or Initiating Event	Importance	Worst Case Importance	Proportional Importance
T_4	.43	.94	.87
.5HR-FTR-OS-PWR	.38	.88	.76
EACPOWER	.28	.78	.55
OPERROR	.28	.78	.55
S/R-VR	.26	.77	.53
IC	.14	.64	.28
FWCI	.11	.62	.22
SWS	.10	.61	.21
RPS	.06	.57	.13
T_2	.06	.57	.13
2HR-FTR-OS-PWR	.05	.56	.11
DCPOWER	.03	.54	.06

Table 116

Millstone
Contribution of Dominant Basic Events, Grouped by Front Line and Support Systems, to Total Core Melt Frequency: Criteria 1 and 2
(Vesely-Fussel Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
T_4	T_4	.43	.94	.87
.5HR-FTR-OS-PWR	.5HR-FTR-OS-PWR	.38	.88	.76
EACPOWER	AC-DGN-LOF	.05	.54	.09
	AC-1514C-1-FTO	.04	.53	.07
	AC-1514A-2-FTO	.04	.53	.07
	AC-14CT-16-FTC	.04	.53	.07
	AC-IV-3-FRC	.04	.50	.01
	AC-F5-1F-FRC	.04	.50	.01
OPERROR	MDP	.28	.78	.55
S/R-VR	S/R-VR	.26	.78	.55
IC	ICM-10-MOV-FTO	.04	.53	.08
	IC-1-MOV-TMC	.02	.51	.04
	IC-2-MOV-TMC	.02	.51	.04
	IC-4-MOV-TMC	.02	.51	.04
	FTR-ICM-10	.01	.50	.03
FWCI	MW-97-SOV-FRO	.01	.51	.03
	AC-14C-3-FRC	.01	.50	.02
	AC-14C-4-FRC	.01	.50	.02
	AC-14A-3-FRC	.01	.50	.02
	AC-E4-3C-FRC	.01	.50	.02
	AC-14C-15-FRC	.01	.50	.02
	AC-E3-5E-FRC	.01	.50	.02
	LCLPS-2-8-TOM	.01	.50	.01
SWS	SW-DGN-HTX-LOF	.06	.55	.12
	SW-448-STR-LOF	.02	.51	.37
	AC-EF7-26-FTC	.01	.50	.25
RPS	A	.06	.57	.13
T_2	T_2	.06	.57	.13
2HR-FTR-OS-PWR	2HR-FTR-OS-PWR	.05	.56	.11
DCPOWER	DC-A-BATT-TMO	.02	.51	.44
	DC-A-BATT-LOF	.01	.50	.16

Table 117

Millstone
Contribution of Front Line and Support Systems to Total Plant Risk: Criterion 3
(Vesely-Fussell Importance Measure)

System, Operator Action or Initiating Event	Importance	Worst Case Importance	Proportional Importance
T_4	.35	.82	.65
EACPOWER	.31	.78	.59
20HR-FTR-OS-PWR	.26	.73	.48
S/R-VR	.21	.68	.40
RPS	.18	.66	.35
T_2	.18	.66	.35
IC	.13	.60	.25
.5HR-FTR-OS-PWR	.09	.56	.17
OPERROR	.05	.52	.09
FWCI	.03	.50	.05
SWS	.02	.50	.47

Table 118

Millstone
Contribution of Dominant Basic Events, Grouped by Front Line and Support Systems, to Total Plant Risk: Criterion 3
(Vesely-Fussel Importance Measure)

System, Operator Action, or Initiating Event	Basic Event	Importance	Worst Case Importance	Proportional Importance
T_4	T_4	.35	.82	.65
EACPOWER	AC-IV-XFR-LOF	.08	.54	.14
	92662-1A-XFO	.05	.48	.08
	92662-1-XFO	.05	.48	.08
	AC-1S14C-1-FTO	.05	.48	.08
	AC-1S14A-2-FTO	.05	.48	.08
	AC-14CT-16-FTC	.05	.48	.08
20HR-FTR-0S-PWR	20HR-FTR-0S-PWR	.26	.73	.48
S/R-VR	S/R-VR	.21	.68	.40
RPS	A	.18	.66	.35
T_2	T_2	.18	.66	.35
IC	FTR-ICM-10	.07	.53	.13
	IC-1-MOV-TMC	.02	.49	.03
	IC-2-MOV-TMC	.02	.49	.03
	IC-4-MOV-TMC	.02	.49	.03
.5HR-FTR-0S-PWR	.5HR-FTR-0S-PWR	.09	.56	.17
OPERROR	MDP	.05	.52	.09
FWCI	MW-97-SOV-FRO	.002	.47	.004
	AC-14C-3-FTC	.002	.47	.004
	AC-14C-4-FRC	.002	.47	.004
	AC-14A-3-FRC	.002	.47	.004
SWS	SW-DGN-HTX-LOF	.02	.49	.03

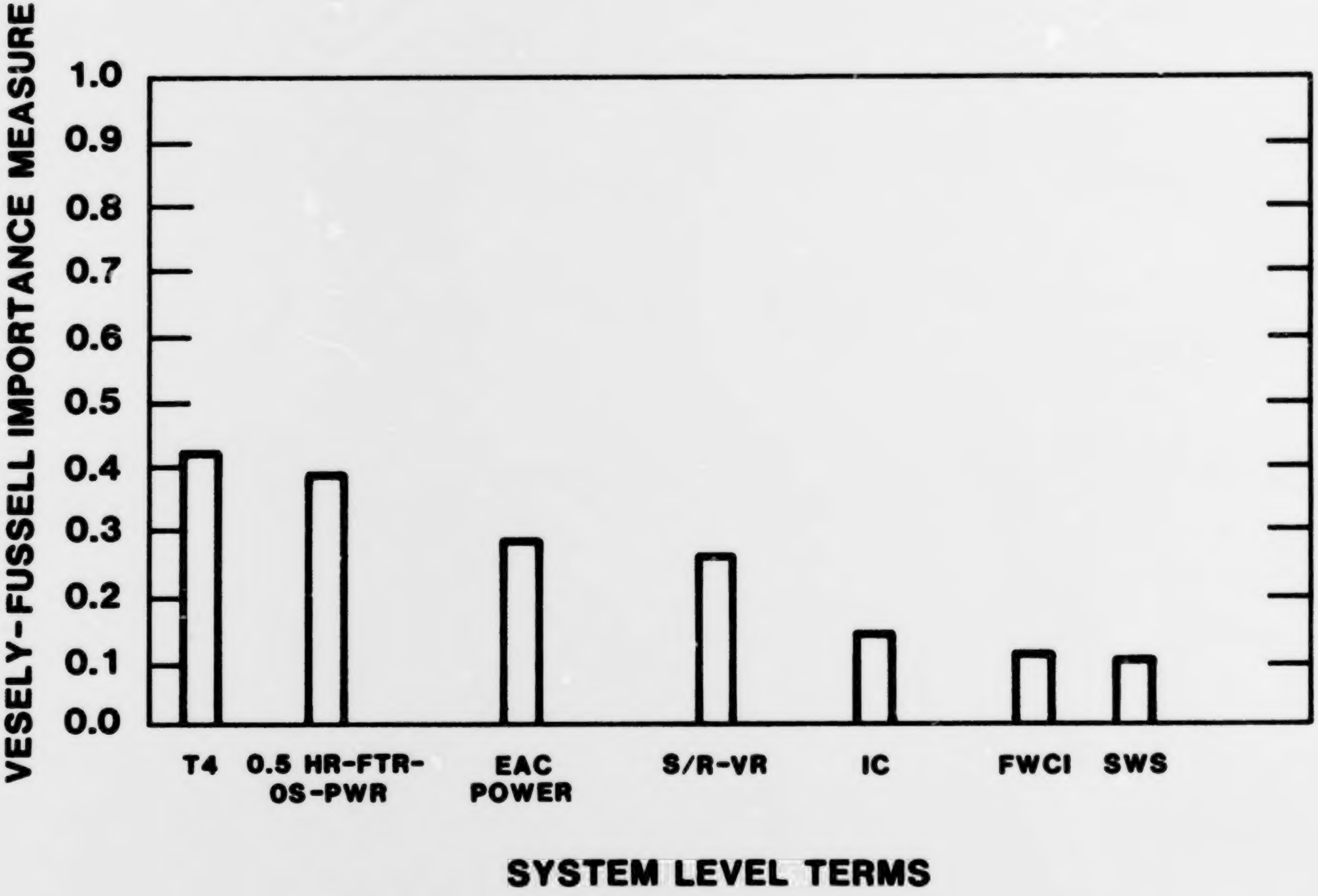

Figure 58. Millstone, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

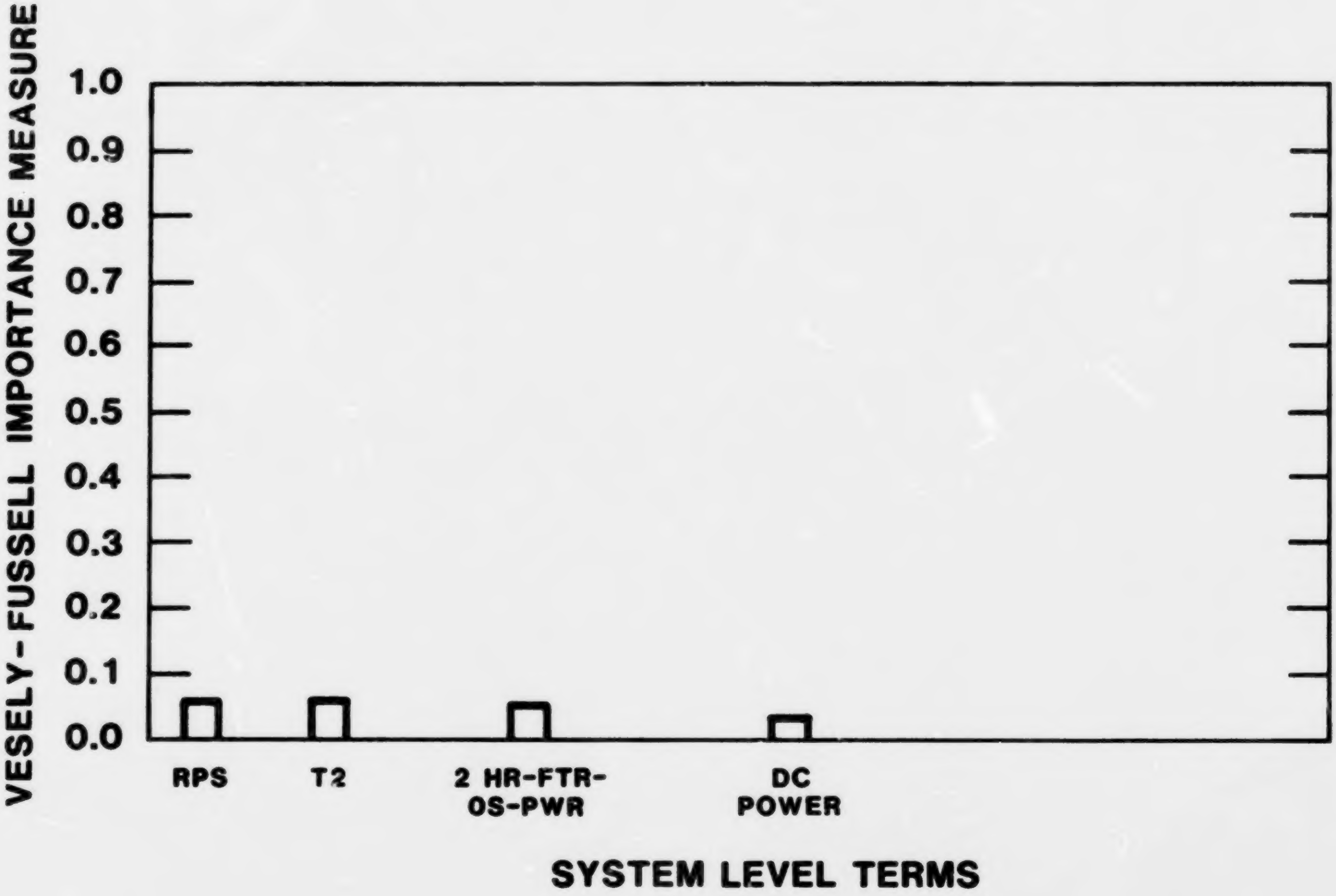

Figure 58. Millstone, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

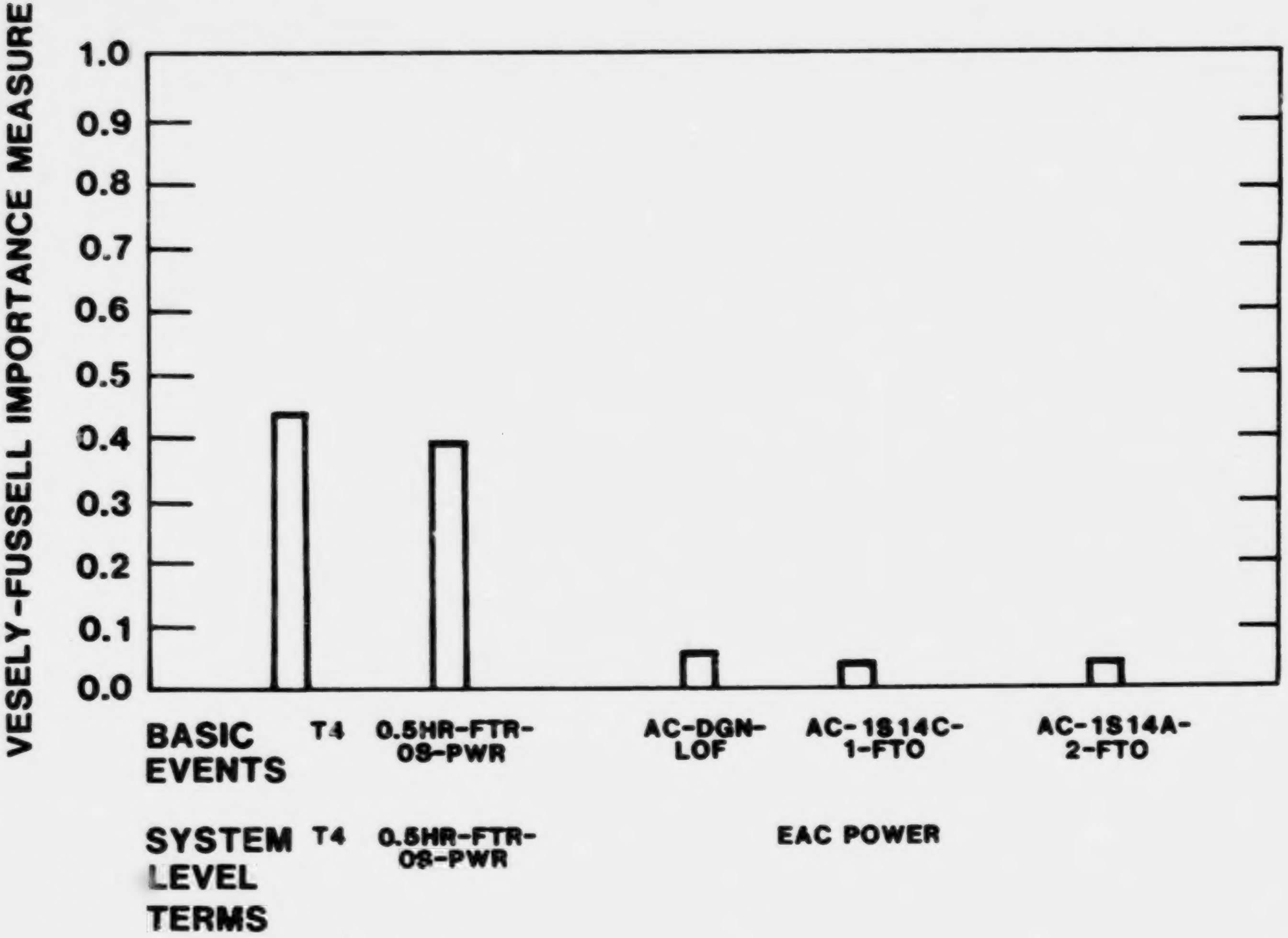

Figure 59. Millstone, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2

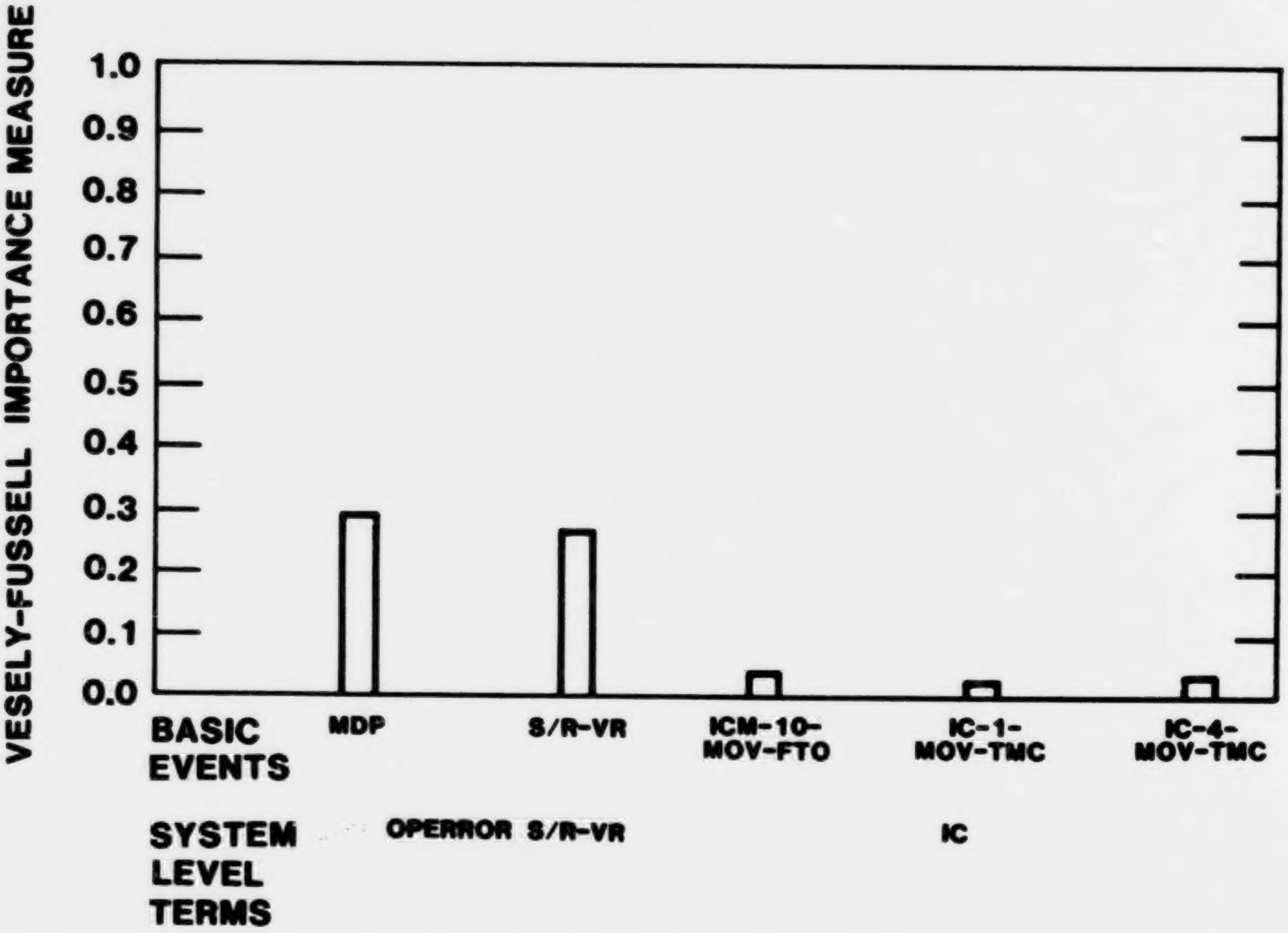

Figure 59. Millstone, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and (Cont)

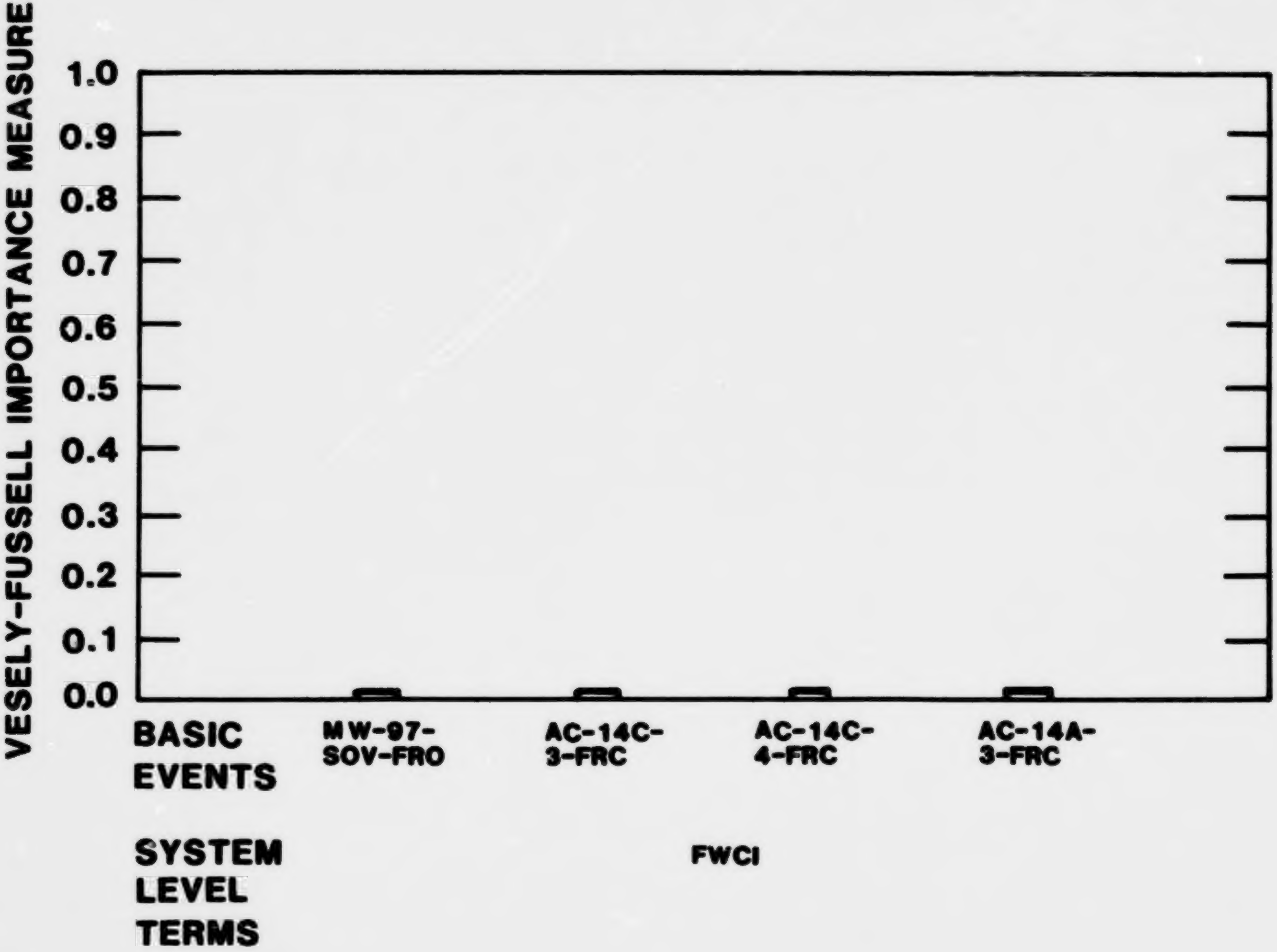

Figure 59. Millstone, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

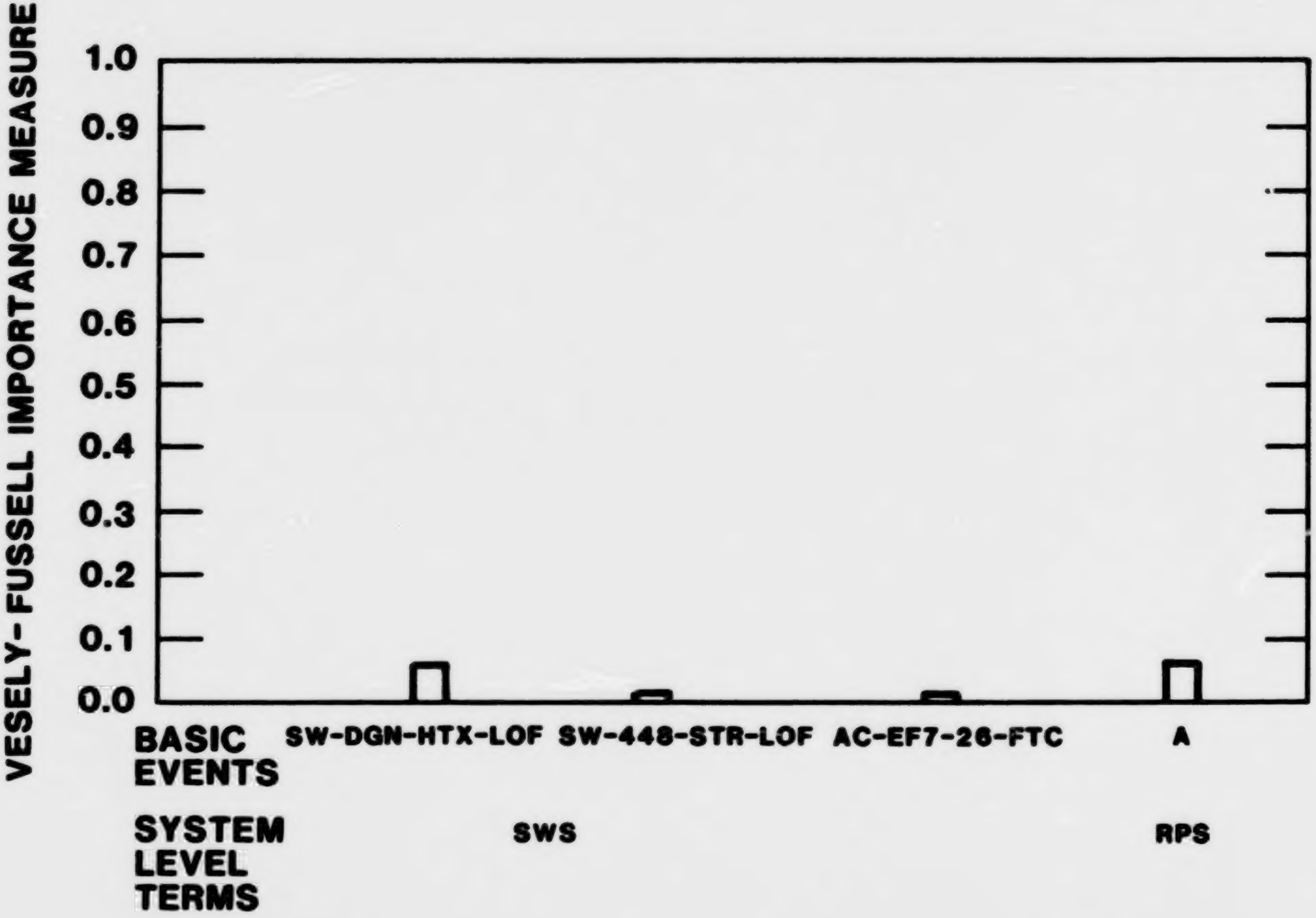

Figure 59. Millstone, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

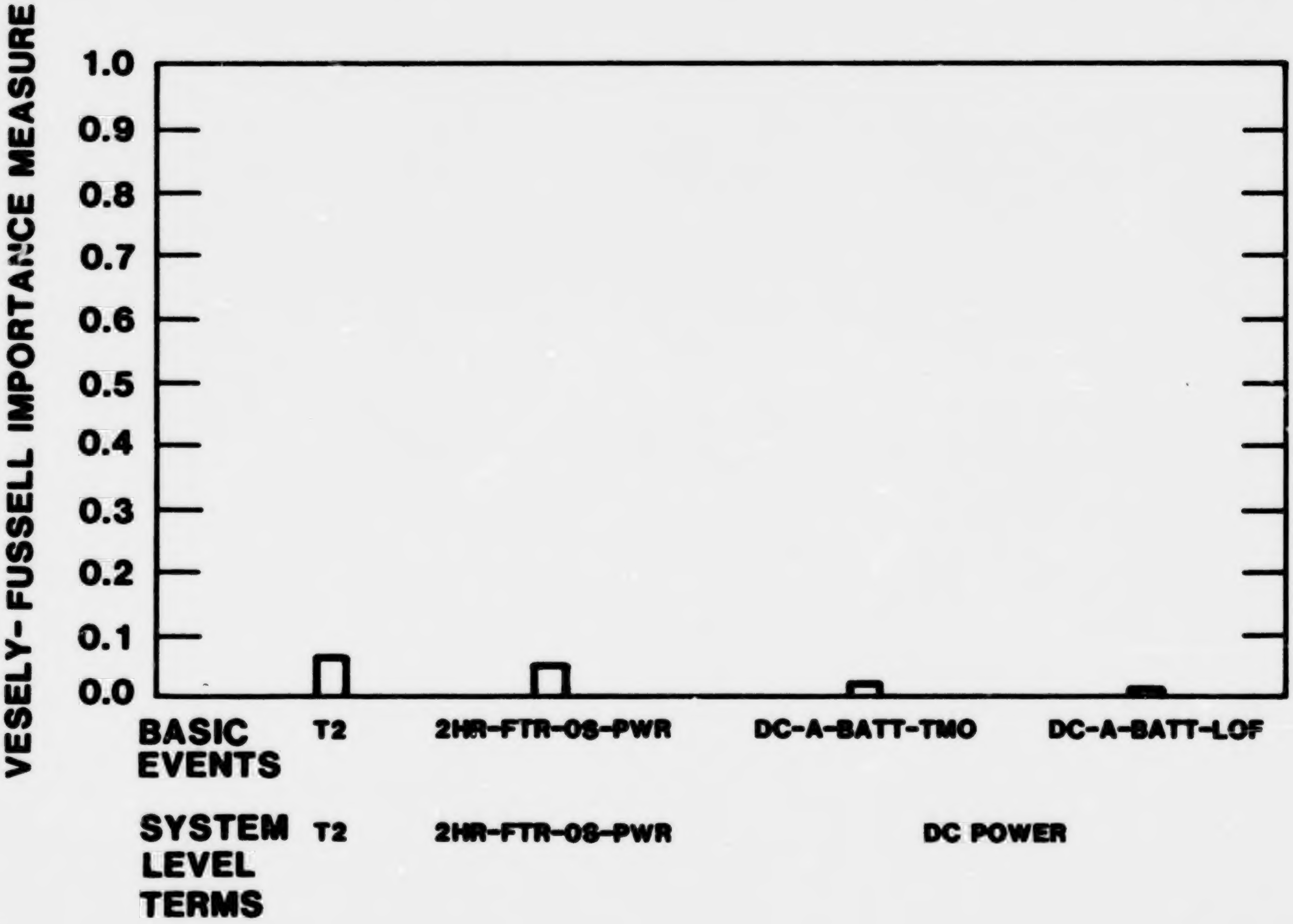

Figure 59. Millstone, Importance of Basic Events, Grouped by Systems, to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

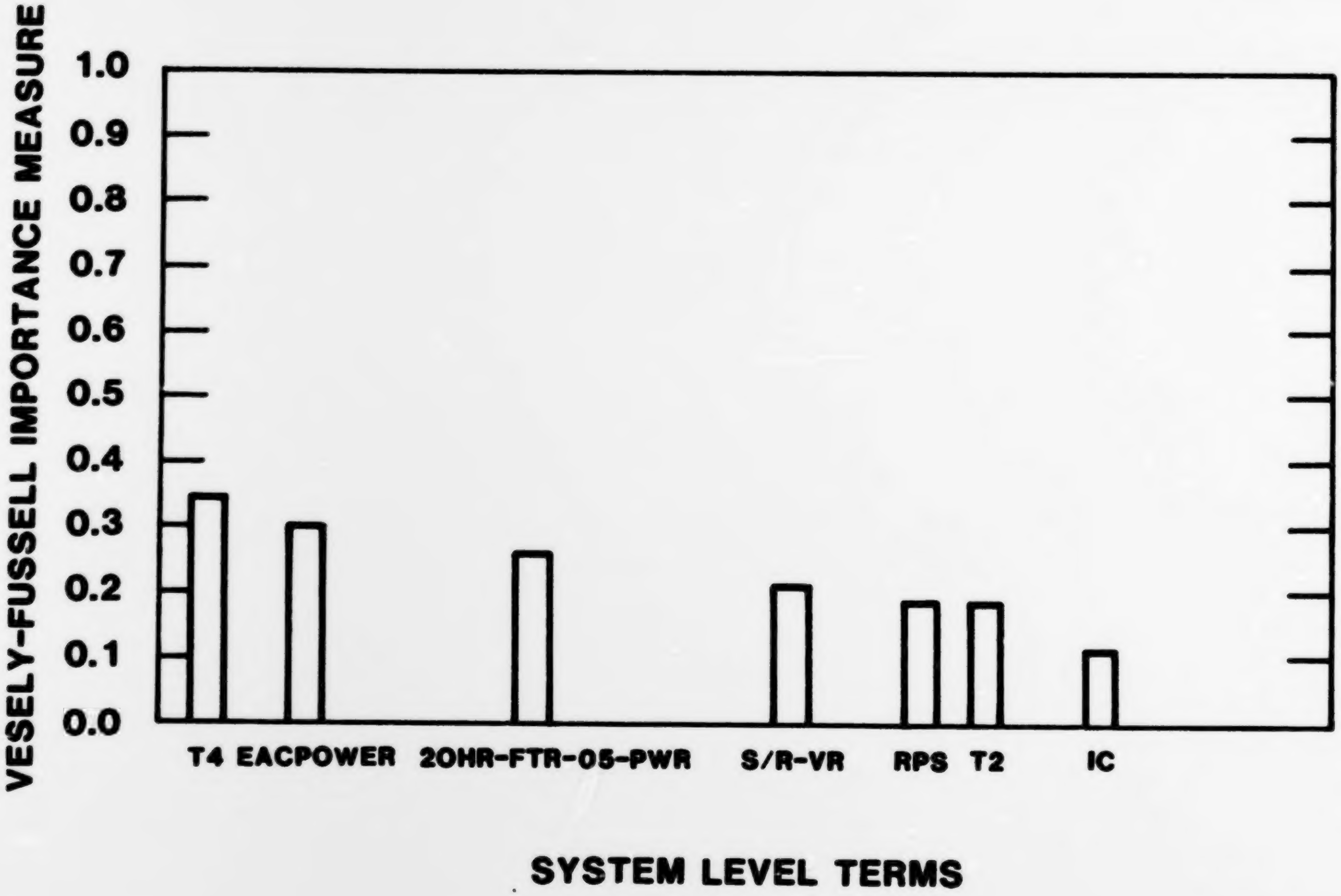

Figure 60. Millstone, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

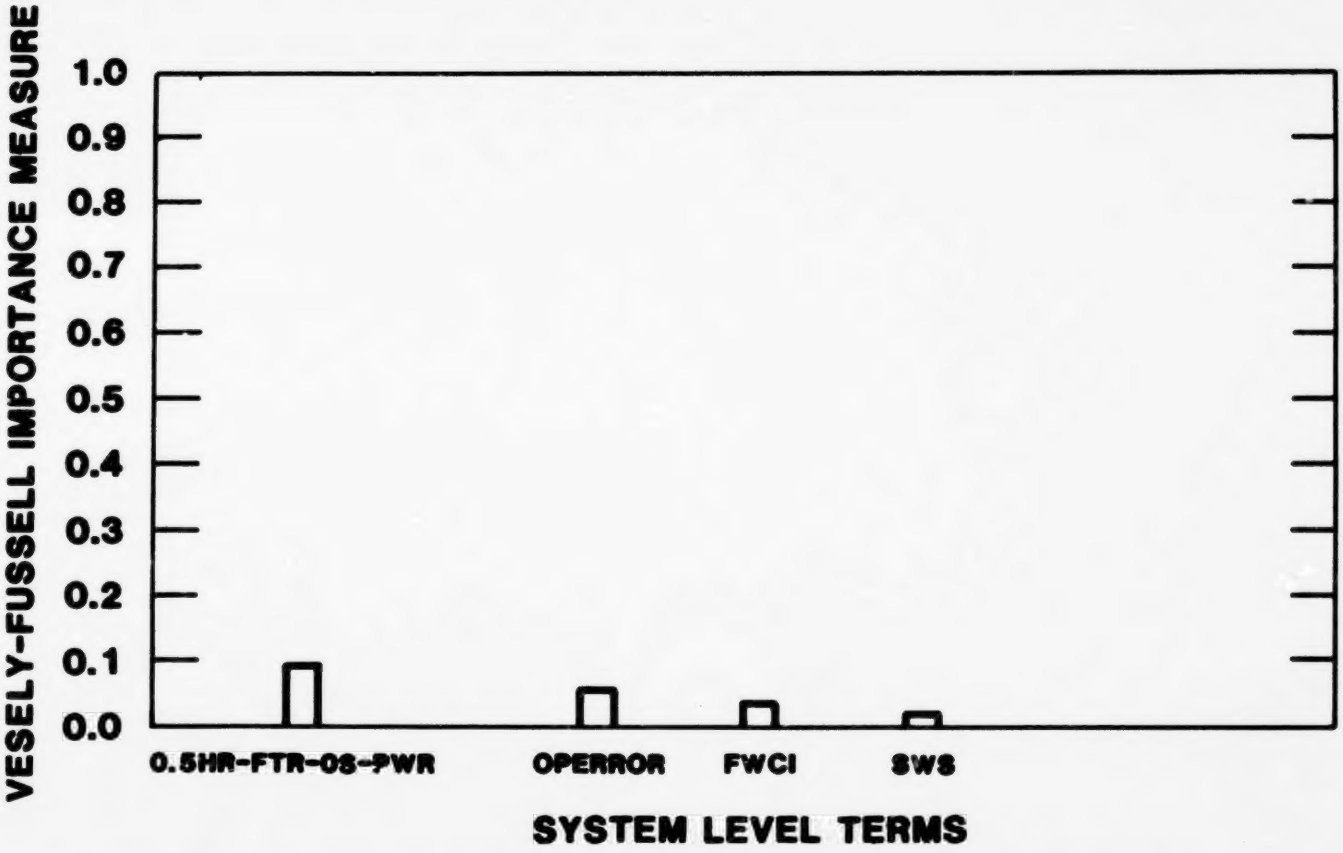

Figure 60. Millstone, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Cont)

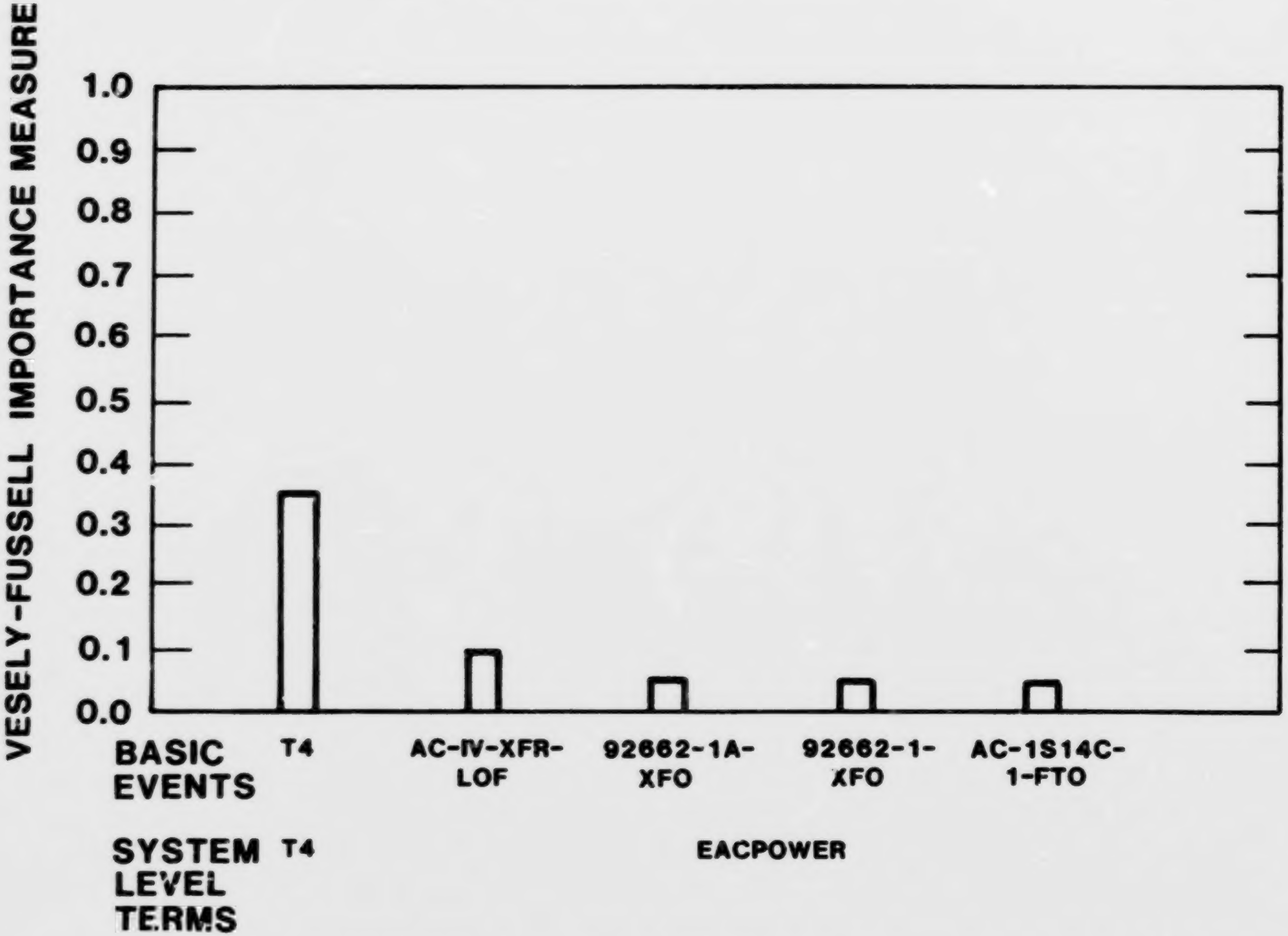

Figure 61. Millstone, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3

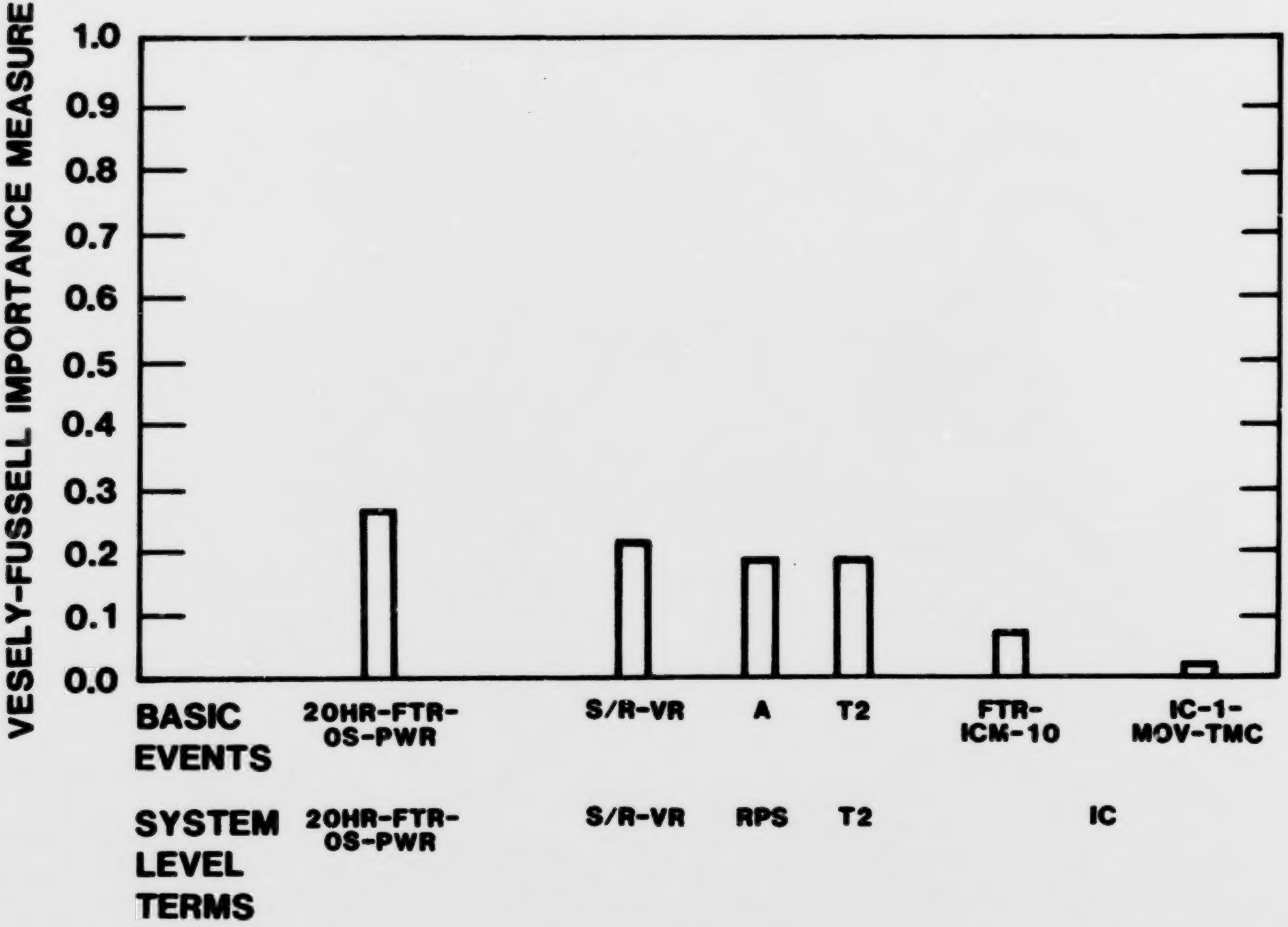

Figure 61. Millstone, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

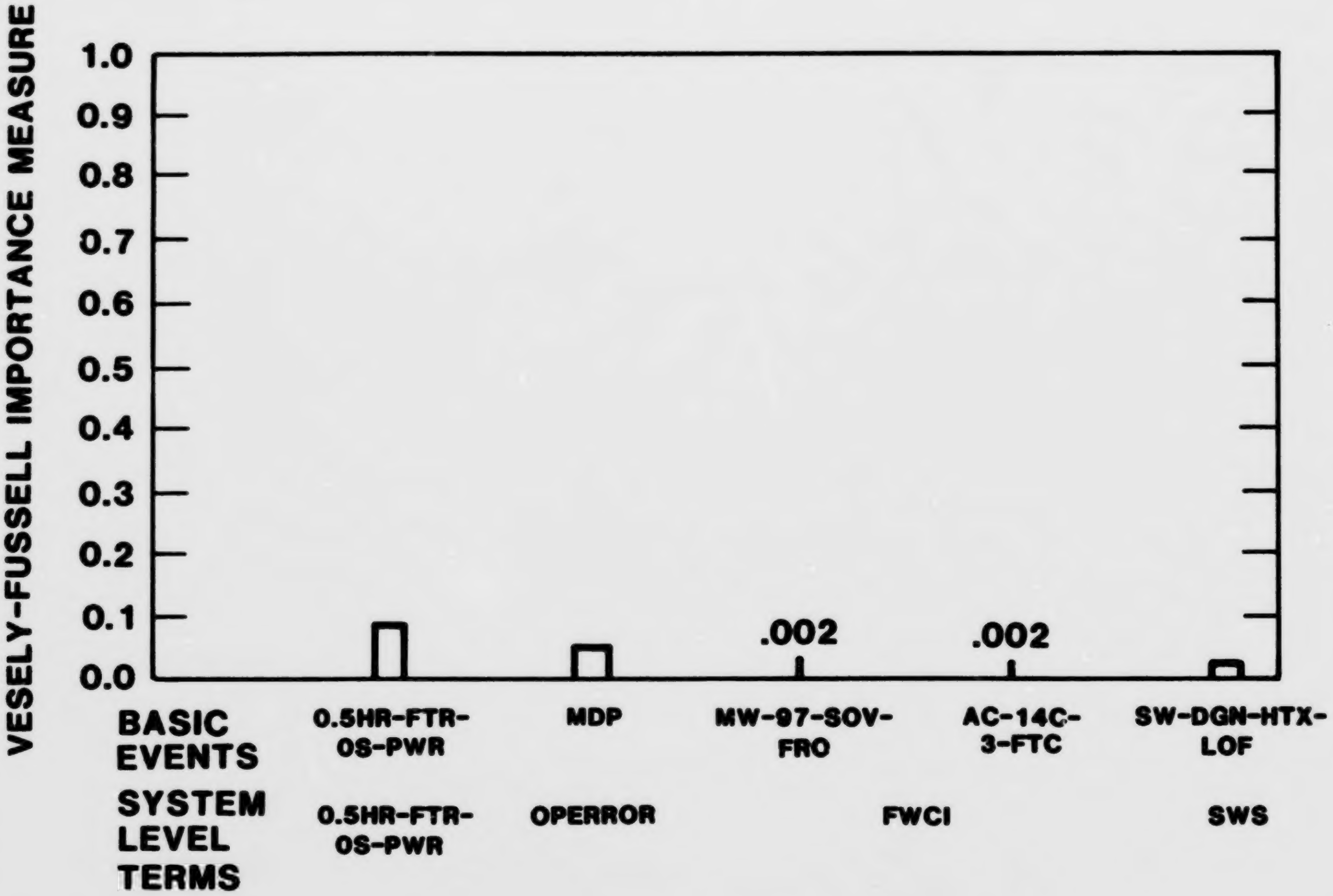

Figure 61. Millstone, Importance of Basic Events, Grouped by Systems, to Total Plant Risk: Criterion 3 (Cont)

C. Inter-Plant Comparisons

1. Westinghouse Plants

Sequoyah - RSSMAP
Surry - WASH-1400
Zion - ZPPS
Indian Point 2 - IPPSS
Indian Point 3 - IPPSS

a. Introduction

All five PRAs performed on Westinghouse plants presented problems for the application of the importance methodology developed for this analysis. The methodology is based on the modeling of core melt sequences by fault trees and Boolean equations. However, none of the PRAs on Westinghouse plants provided any such equations used in quantifying dominant sequences. The cut sets for the importance analysis for these PRAs were developed using the event trees and event descriptions. The results for all five PRAs are very general basic events comprising the cut sets. Such a crude level of analysis may fail to account for common failures between systems within an accident sequence. The "resolution" of such analysis is also very poor. Importance calculations may only be for a certain train or loop of a system, or even for the system itself. The importance of an actual hardware component or human action causing a system to fail is lost by the general nature of the simplified sequence equations. This is illustrated by the very close, almost one-to-one correlation of system level terms to basic event terms for each of the Westinghouse plants discussed in Section III.A-1. The results for the Westinghouse plants should be interpreted cautiously as the quantitative utility of the importance measure is limited due to the general nature of the cut sets developed from the PRAs. The importance calculations are useful for studying the plants from a qualitative, macroscopic viewpoint, and for understanding the relative importance of plant systems to one another.

The list below shows the percent of core melt frequency modeled by the cut sets developed for each Westinghouse plant:

Plant	Percent Core Melt Frequency Modeled
Sequoyah	88%
Surry	94%
Zion	93%
Indian Point 2	84%
Indian Point 3	85%

Clearly, the simplified fault trees and the resulting cut sets model a very significant portion of the total core melt frequency for each plant. The major gap between total core melt frequency and that portion represented by the cut sets is due to the elimination of all sequences contributing less than 5 percent to total core melt frequency from the analysis. However, for some sequences, it was not possible to account for all of the frequency of the sequence with the simplified fault trees derived from the sequence descriptions in the PRAs. This adds to the modeling gap as well as the elimination of nondominant sequences. The problem in not being able to reproduce the total frequency for some sequences of the PRAs represents the difficulty often encountered in trying to reproduce the results of PRAs using only the information and models published in the PRAs. However, this was not a serious problem with the PRAs on Westinghouse plants. It is not surprising that the cut sets for the PRAs model a very large percent of total core melt frequency, since the cut sets are developed virtually at the event tree level.

b. LOCA Initiating Events

Table 119 lists a summary for the various initiating events covered by the various PRAs on Westinghouse plants. The size limitations for small, medium, and large LOCAs are the limits used in WASH-1400. The various LOCA events for the other PRAs can be closely approximated into these limits.

The Sequoyah PRA used the same methodology as WASH-1400 to calculate LOCA frequencies, using pipe and valve rupture failure rates. The Sequoyah LOCA rates match very closely to the Surry rates. The slightly higher S_1 and S_2 Sequoyah rates are attributed in the PRA to the fact that Sequoyah in a four loop RCS plant and Surry in a three loop RCS plant.[7] However, the large LOCA frequency for Sequoyah is significantly less than that for Surry. This does not follow the reasoning given for Sequoyah S_1 and S_2 frequencies. Sequoyah was the earliest RSSMAP study, and in the other RSSMAP studies (Grand Gulf, Calvert Cliffs, and Oconee) LOCA frequencies were based more directly on WASH-1400 values.

The Zion and Indian Point LOCA frequencies are higher than those for Sequoyah and Surry. This is attributed to the Bayesian methodology used in the Zion and Indian Point studies to establish probability distributions.[12] This methodology tends to make the LOCA frequencies larger than the methodology of WASH-1400.

The Interfacing Systems LOCA frequencies for Surry and Sequoyah are very similar. In both plants, the dominant

Interfacing System LOCA is the failure of check valves separating the RCS from the LPIS to reseat after LPIS flow tests. The slightly higher value of this event, event V, for Sequoyah is due to differences in plant configurations according to the Sequoyah PRA. But the difference is not significant to this analysis. Event V frequencies for Zion and Indian Point are significantly lower than Surry and Sequoyah. For the Zion and Indian Point paths, the dominant interfacing systems are the RCS and the RHRS. The increased testing frequency of LPIS check valves at Zion and Indian Point is attributed for this.[12] Thus, for Zion and Indian Point plants, sequence (and event) V is far less frequent than the dominant sequences by two to three orders of magnitude. Yet for Surry and Sequoyah, where the LPIS/RCS interface is dominant, sequence V is a significant contributor to core melt frequency, and even more significant to risk, especially for Surry (Table 21). Due to the single event nature of sequence V, the importance measure for this event is a very accurate representation of the event's impact on core melt frequency and risk, despite the generalized nature of the cut sets developed for Westinghouse plants.

c. Transient Events

The transient event frequencies for the various plants reflect plant specific experience. Although some differences do exist from plant to plant, the differences seem reasonable since plant specific information is used.

d. External Events

Only the Zion and Indian Point studies considered external vents as possible initiators for core melt sequences, so it is impossible to draw comparisons across all Westinghouse plants. Even across the four plants for which external events have been considered (Zion 1 and 2, IP 2 and 3) significant differences reflect plant and site specific characteristics. The threat of fire at both Zion and Indian Point is primarily centered around fire in the cable spreading rooms and cable tunnels. Yet, plant specific differences yield a significant range of frequencies at the various units. Fire sequences are not dominant at Zion, but are significant at Indian Point. Yet, even at Indian Point, almost an order of magnitude exists between IP2 and IP3. An example of plant specific design which can impact fire sequences is the layout of equipment. At IP3, the diesel generators are in separate rooms, while at IP2 the diesel generators are all in the same room. The impact of seismic events upon plants is also very site and design dependent. Zion and Indian Point 3 have similar core melt frequencies due to seismic events, but Indian Point 2 has a significantly higher frequency. Indian Point 2 is located close enough to Indian Point 1, a fossil fuel unit, that the collapse of unit 1's superheater smoke stack onto unit 2's control room adds a dimension to seismic events for Indian Point 2 not applicable to Indian Point 3 and Zion. Thus, the dominance of external events to core melt accidents appears to be very site and plant dependent. Fire sequences are dominant at both Indian Point 1 and 2, but not Zion. Seismic sequences are dominant at Zion and Indian Point 2, but not Indian Point 3.

e. Containment Response

The Zion and Indian Point studies have significant differences for results of Criteria 1, 2, and 3. For both studies, among those sequences representing 5 percent or more of the criterion frequency, only external event sequences result in containment failure, whereas in the Sequoyah and Surry PRAs, all core melt sequences result in containment failure. A detailed discussion of containment response methodologies is far beyond the scope of this analysis. However, significant differences in the methodology used for Sequoyah and Surry, and the methodology used for Zion and Indian Point have resulted in very different containment core melt sequence release probabilities. The Sequoyah and Surry containment failure models used assumptions of such factors as containment failure pressure, amount of cladding removed from fuel, and quantity of hydrogen generated. The Zion and Indian Point studies used a methodology employing more detailed structural and chemical analyses of containment and fuel response to accidents than did Surry and Sequoyah. Certain containment response parameters such as failure pressure and hydrogen generation can vary greatly between the two methodologies.

f. Dominant System and Basic Events

The system level results for the Westinghouse plants are reproduced here in Figures 62 through 74. The results show a significant impact upon core melt accidents by operator errors. Operator failures to properly manage switchover to recirculation during LOCA sequences are the most significant operator errors at Sequoyah, Surry, Indian Point 3, and Zion. At Indian Point 3 and Zion, operator failure to switchover to recirculation during large LOCAs is the single most important operator error. The frequencies of operator errors during large LOCAs are higher than the frequencies of operator errors during small LOCAs. This is to account for the high stress environment among operators during large LOCAs. No large LOCA sequences are dominant at Sequoyah and Surry. The frequency of large LOCAs for Zion and Indian Point 3 is significantly higher than for Surry and Sequoyah.

Only at Indian Point 2 are no operator errors important to core melt accidents. None of the dominant accident sequences at IP2 analyzed in this report involve recirculation failures, either from hardware or human failures. This is not due to a higher level of competence among Indian Point 2 operators or a better recirculation system than at other plants. Inspection of Tables 8.3-9 and 8.3-10 of the IPPSS show almost equivalent frequencies between Indian Point 2 and 3 for sequences involving failure of recirculation. The frequencies of human failures to initiate recirculation at Indian Point 2 and 3 are also equal (Tables 1.5.2.3.4-13 and 1.8.2.3.4-13 of the IPPSS). Indian Point 2 sequences involving system failures are completely dominated by core melt sequences due to external events (Table 31). Indian Point 2 sequence SEFCTT, loss of offsite power and failure of emergency AC power, is the only nonexternal event sequence.

Sequoyah has another human error besides failure to initiate recirculation which is important to core melt accidents. This is operator error CMSD (Table 10). Sequoyah is the only Westinghouse plant analyzed which has an ice condenser containment. Ice condenser containments have an upper and lower containment connected by drains. Failure to open these drains after refueling would fail both CSRS and ECRS during a recirculation sequence.

Emergency AC power is important to core melt accidents only at Surry and Indian Point 2. Thus, only for these two plants do loss of emergency AC power sequences dominate core melt frequency, TMLB' for Surry, SEFCTT (Sequence 4 on Table 8.3-9 of the IPPSS) for Indian Point 2. The Sequoyah PRA attributes a better availability of the diesel generators at Sequoyah than at Surry to load sequencing.[7] The diesel generators at Sequoyah are subjected to a significantly less startup load than Surry's, improving the startup availability of Sequoyah diesel generators. At Indian Point 3, all three diesels must fail before the emergency AC system fails to power such vital support systems as the Component Cooling Water System (CCWS) (Section 1.6.2.3.7 of the IPPSS). At Zion, Units 1 and 2 share five CCWS pumps. Each pump has a separate diesel generator dedicated to it (Section 1.5.2.3.7, Zion PSS). At Indian Point 2, only two diesels are dedicated to the CCWS (Section 1.5.2.3.7, IPPSS), resulting in a significantly higher failure rate of the emergency AC power mission than at Indian Point 3 and Zion. Plant specific design appears to impact significantly on the unavailability of emergency AC power.

Safety injection systems are significant only at Sequoyah and Surry. The frequencies of sequences involving failure of High Pressure Injection System (HPIS) or Low

Pressure Injection System (LPIS) at all five plants are fairly equivalent. This suggests that the safety injection systems at the various plants have similar unavailabilities for similar sequences. However, sequences with safety injection failures at Zion and Indian Point 2 and 3 are dominated by recirculation and external event sequences.

Recirculation systems failures are significant at all the plants but Indian Point 2. As discussed above earlier in this section, sequences with recirculation failures at Indian Point 2 do not dominate. However, Indian Point 2 recirculation failures are similar to those at Indian Point 3 and Zion. The IPPSS and Zion PSS models HPRS, LPRS, and RHRS as one system, the Recirculation System, so division of recirculation failures into High, Low Pressure, or Residual heat removal modes was not as clear as it was for Sequoyah and Surry. Several failures of recirculation are discussed in each PRA. However, recirculation failures tend to be dominated by single failures in the charging or safety injection pumps and lines of the recirculation systems.

g. Summary

The results for the five Westinghouse plants indicate that plant specific designs can greatly impact core melt accidents. This is illustrated by the different importance of emergency AC power among the plants, and the significant importance of seismic and hurricane events at Indian Point 2 due to its proximity to the Indian Point 1 superheater stack.

Two important features of the PRAs analyzed here should be kept in mind when evaluating the results of this analysis. First, cross comparisons of results between the various plants can be difficult because three different styles of PRA are observed here, WASH-1400, RSSMAP, and that of the Zion PSS and IPPSS. Sequences, systems, and quantification of unavailabilities differ among the three styles. The presentation of information and results differ as well. A great deal of familiarity with a particular PRA is often required to extract information regarding sequences and system failures. Secondly, none of the PRAs analyzed here modeled core melt sequences with sequence Boolean equations, or at least such models were not published in the PRA documents. Yet, the methodology for calculating the Vesely-Fussel importance measure for basic events is based on the use of cut sets as defined by sequence Boolean equations.

Table 119

Frequency of Initiating Events (R-yr.)$^{-1}$
Westinghouse Plants

	Sequoyah		Surry		Zion	
Initiating Event	Initiating Event	Frequency	Initiating Event	Frequency	Initiating Event	Frequency
1. Large LOCA 6"	A	4.7E-5	A	1.0E-4	A	9.4E-4
2. Medium LOCA 2" 6"	S_1	9.8E-4	S_1	3.0E-4	M	9.4E-4
3. Small LOCA 2"	S_2	1.8E-3	S_2	1.0E-3	S	3.5E-4
4. Interfacing Systems LOCA	V	4.6E-6	V	4.0E-6	V	1.0E-7
5. Transient Loss of Offsite Pwr	T_1	0.2	TFW*RLOP	.2	Transient-LOP	5.8E-2
6. Transient Loss of Main Feedwater	T_2	3.0	TFW	3.0	Loss of MFWS	5.2
7. Other Transients	T_3	4.0	All Other Transients	7.0	All Other Transients	7.5
8. External Events					Seismic Event	5.6E-6
					Fire	E-6

Table 119 (Cont.)

Frequency of Initiating Events $(R\text{-}yr.)^{-1}$
Westinghouse Plants

Initiating Event	Indian Point 2 Initiating Event	Frequency	Indian Point 3 Initiating Event	Frequency
1. Large LOCA 6"	A	1.9E-3	LLOCA	2.0E-3
2. Medium LOCA 2" 6"	M	1.9E-3	MLOCA	2.0E-3
3. Small LOCA 2"	S	1.8E-3	SLOCA	2.0E-2
4. Interfacing Systems LOCA	V	4.7E-7	V	4.6E-7
5. Transient Loss of Offsite Pwr	TT	.2		.3
6. Transient Loss of Main Feedwater		6.7		3.8
7. Other Transients		14.1		5.7
8. External Events	Fire Hurricane Seismic Events	1.9E 2.7E-5 1.4E-4	Fire Hurricane Seismic Events	6.0E-5 0.0 2.4E-6

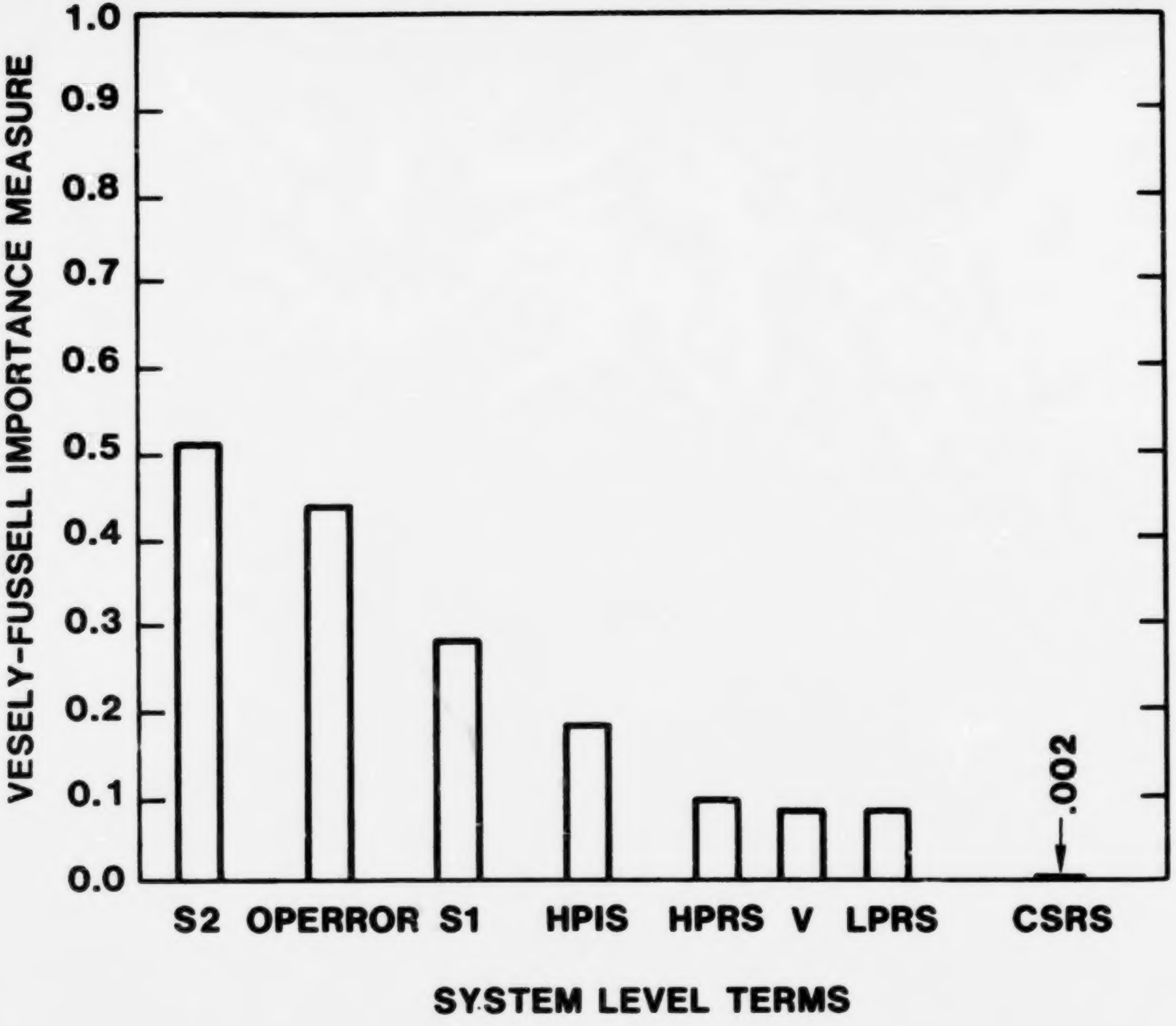

Figure 62. Sequoyah-1, Front Line and Support Systems Importance to Total Core Melt: Criteria 1 and 2

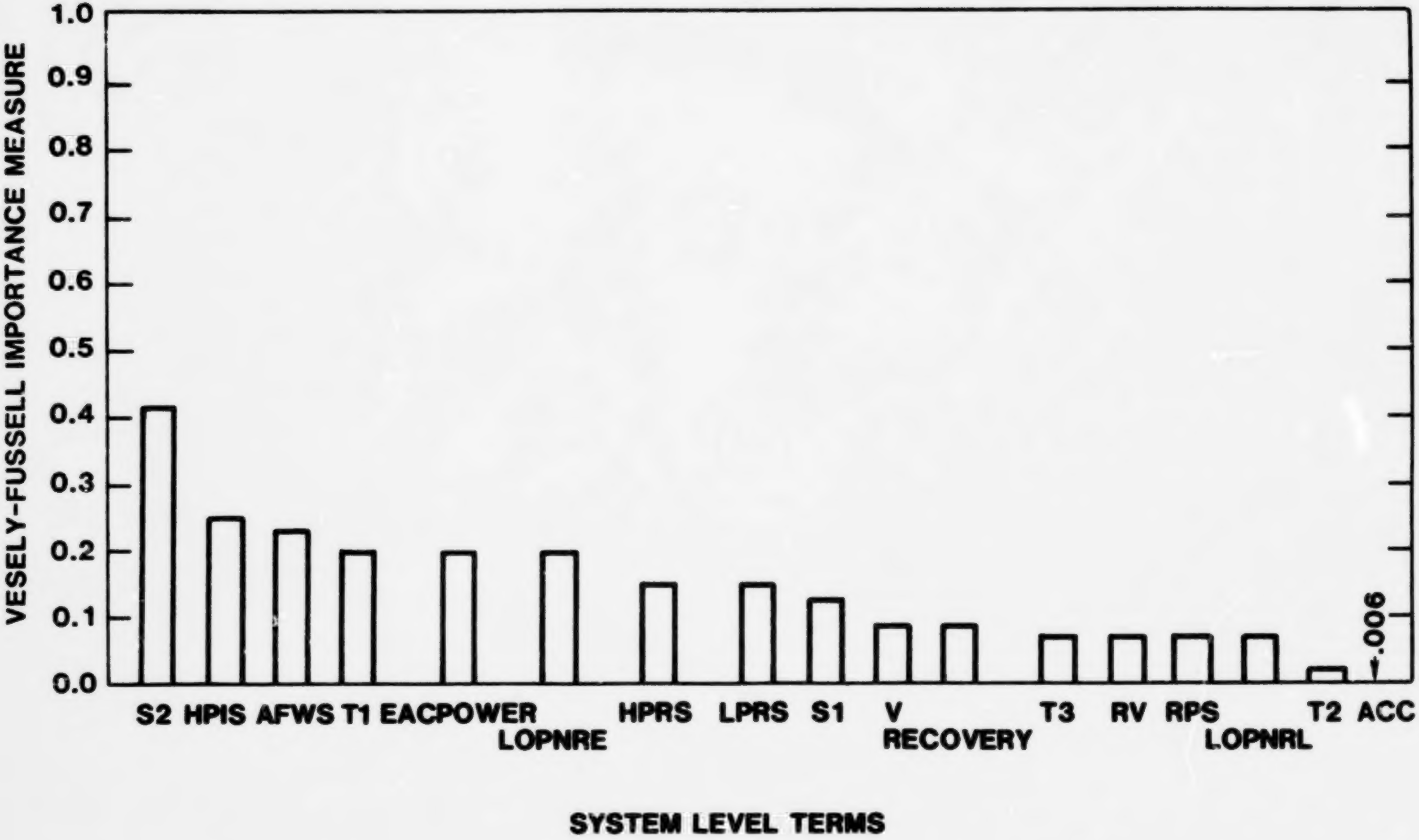

Figure 63. Surry, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

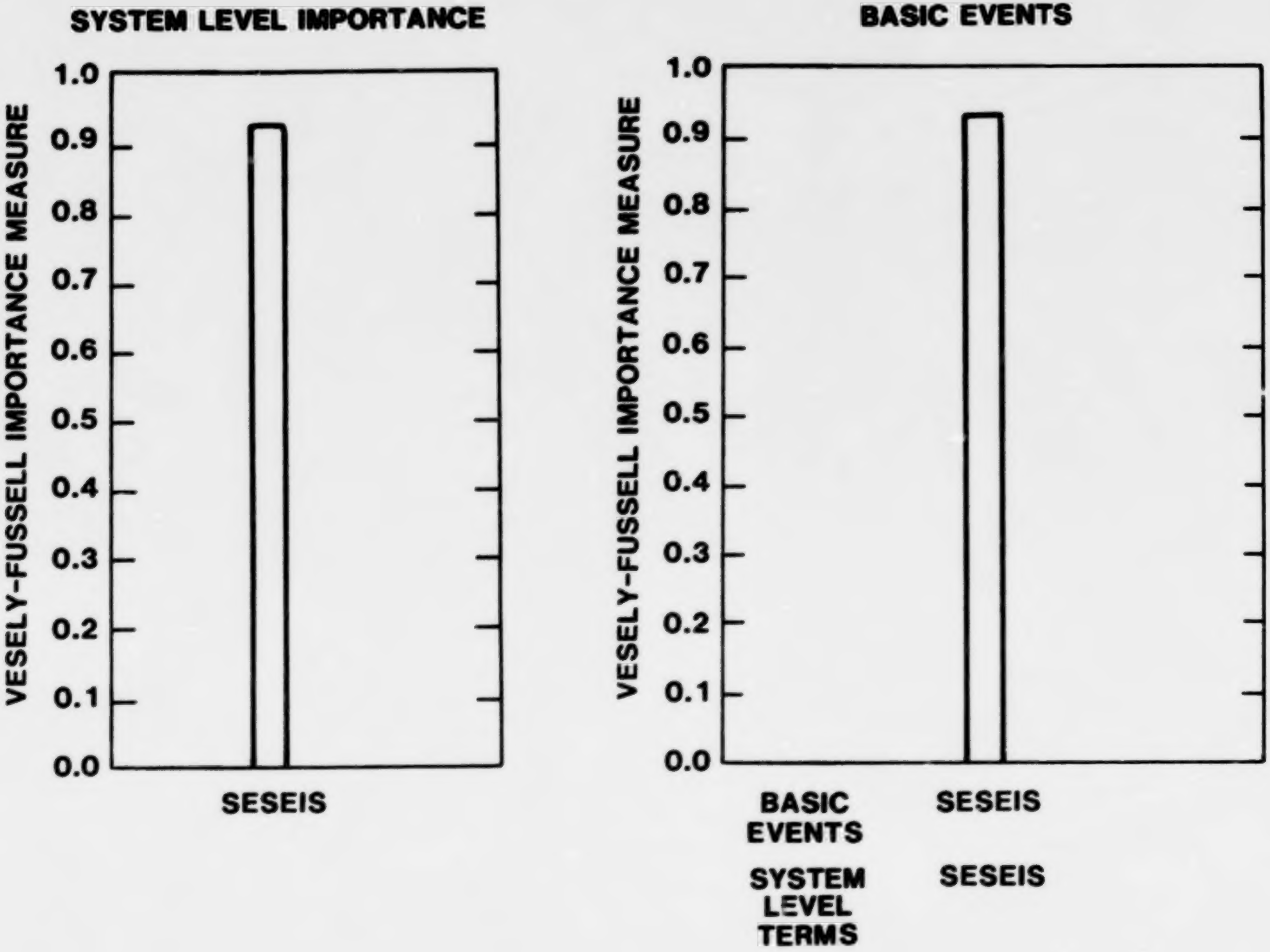

Figure 64. Zion, Importance of Front Line and Support Systems, and Basic Events to Total Core Melt Frequency with Release: Criterion 1

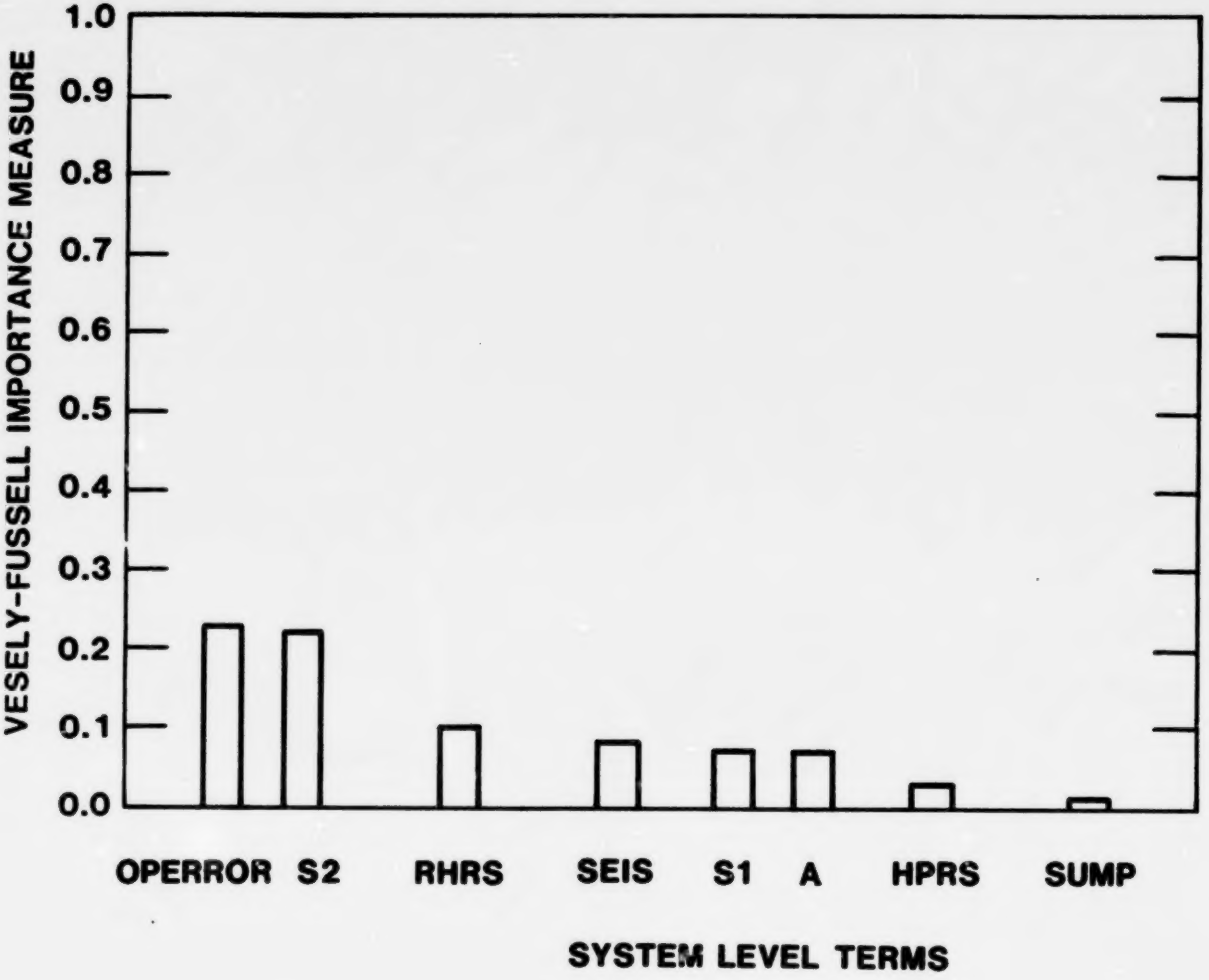

Figure 65. Zion, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2

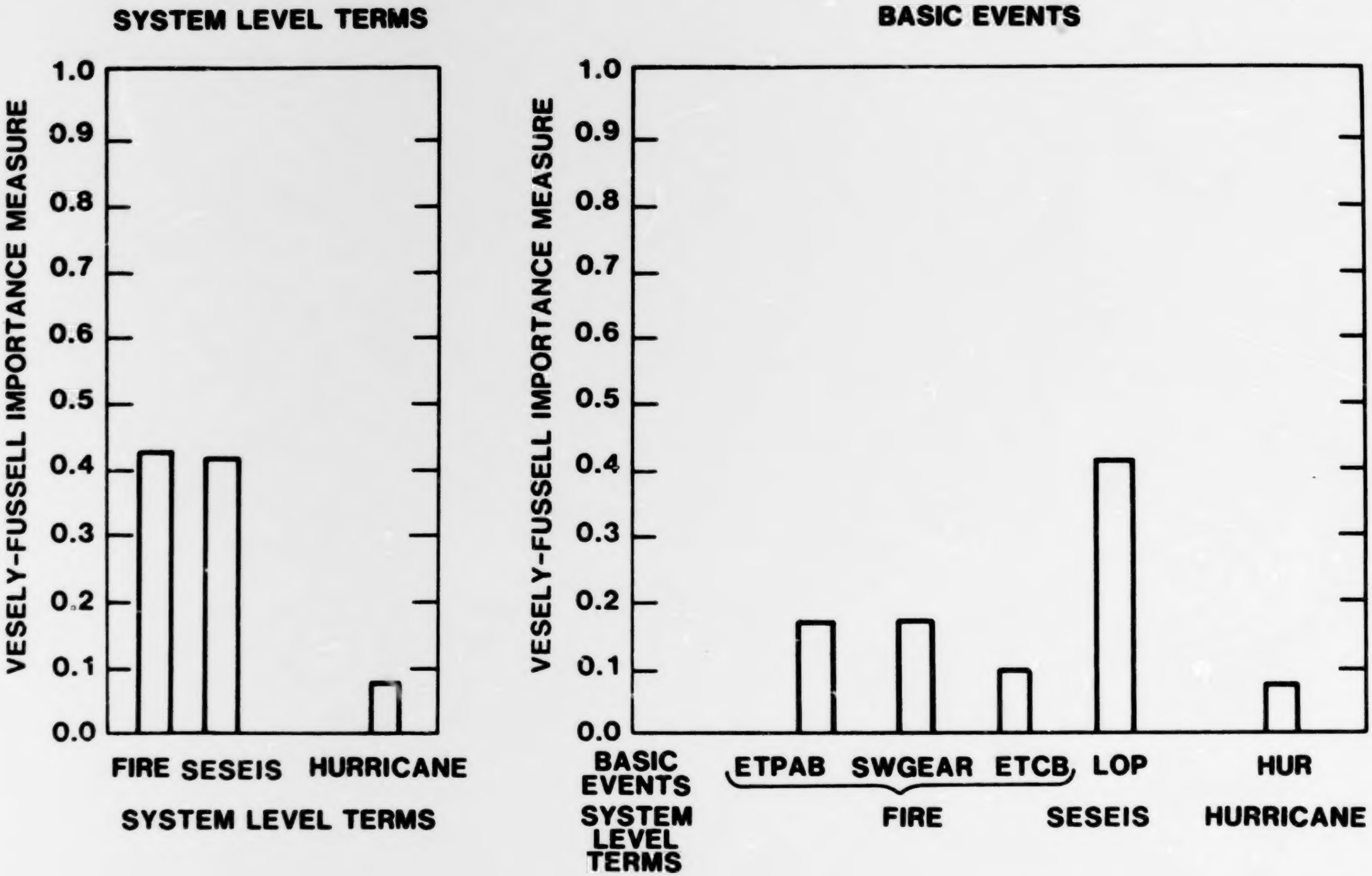

Figure 66. Indian Point 2, Importance of Front Line and Support Systems, and Basic Events, to Total Core Melt Frequency with Release: Criterion 1

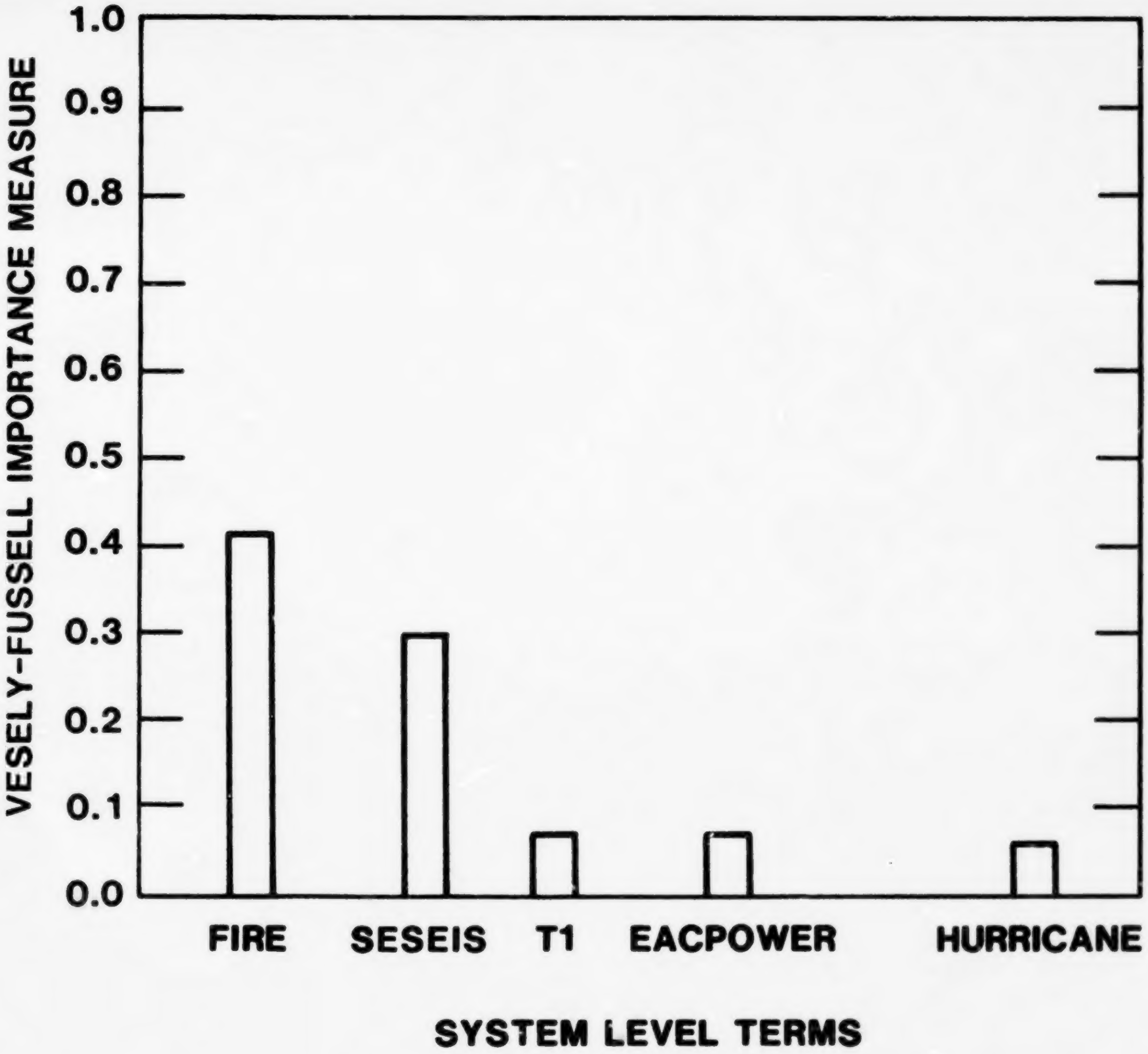

Figure 67. Indian Point 2, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2

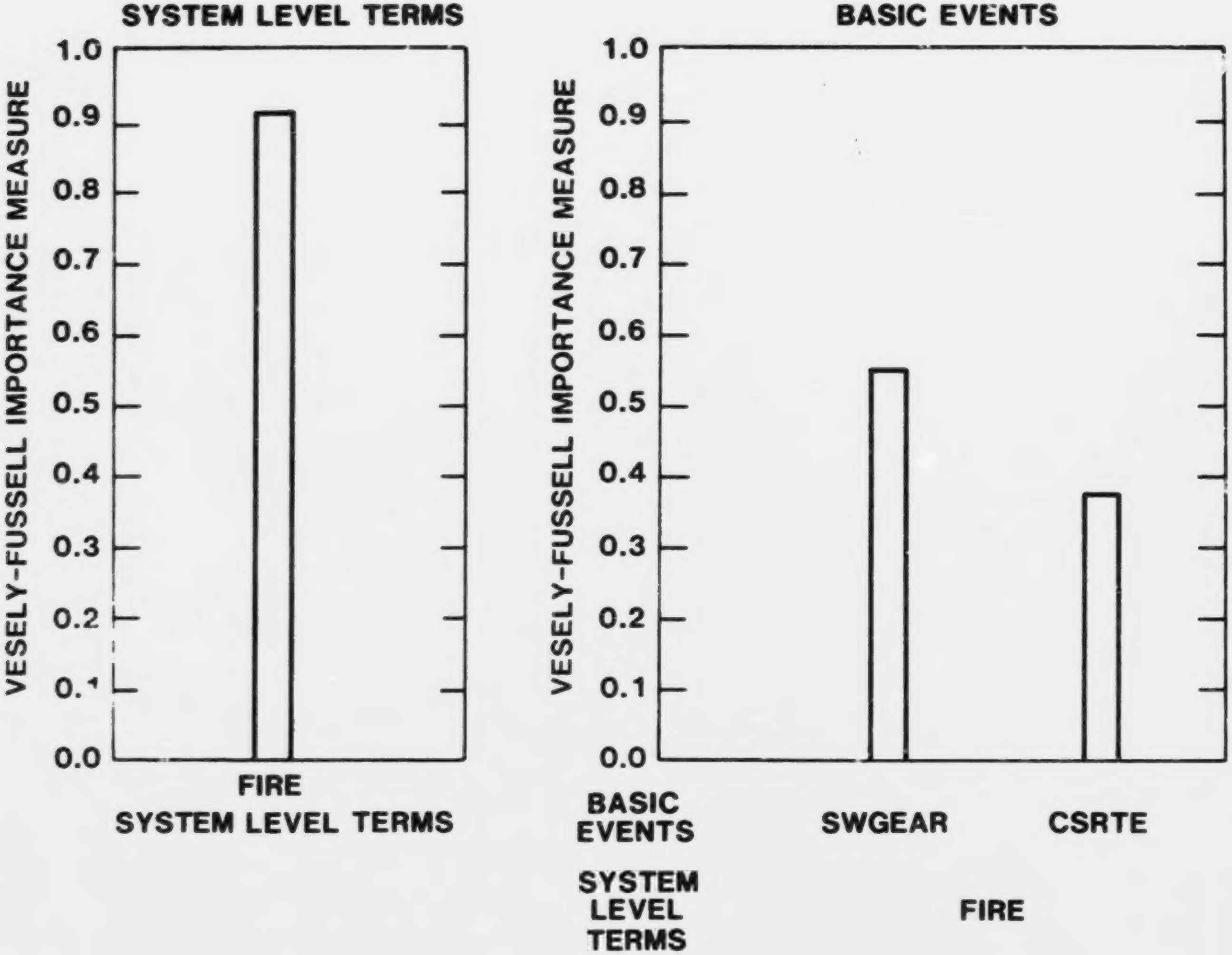

Figure 68. Indian Point 3, Importance of Front Line and Support Systems, and Basic Events, to Total Core Melt Frequency with Release: Criterion 1

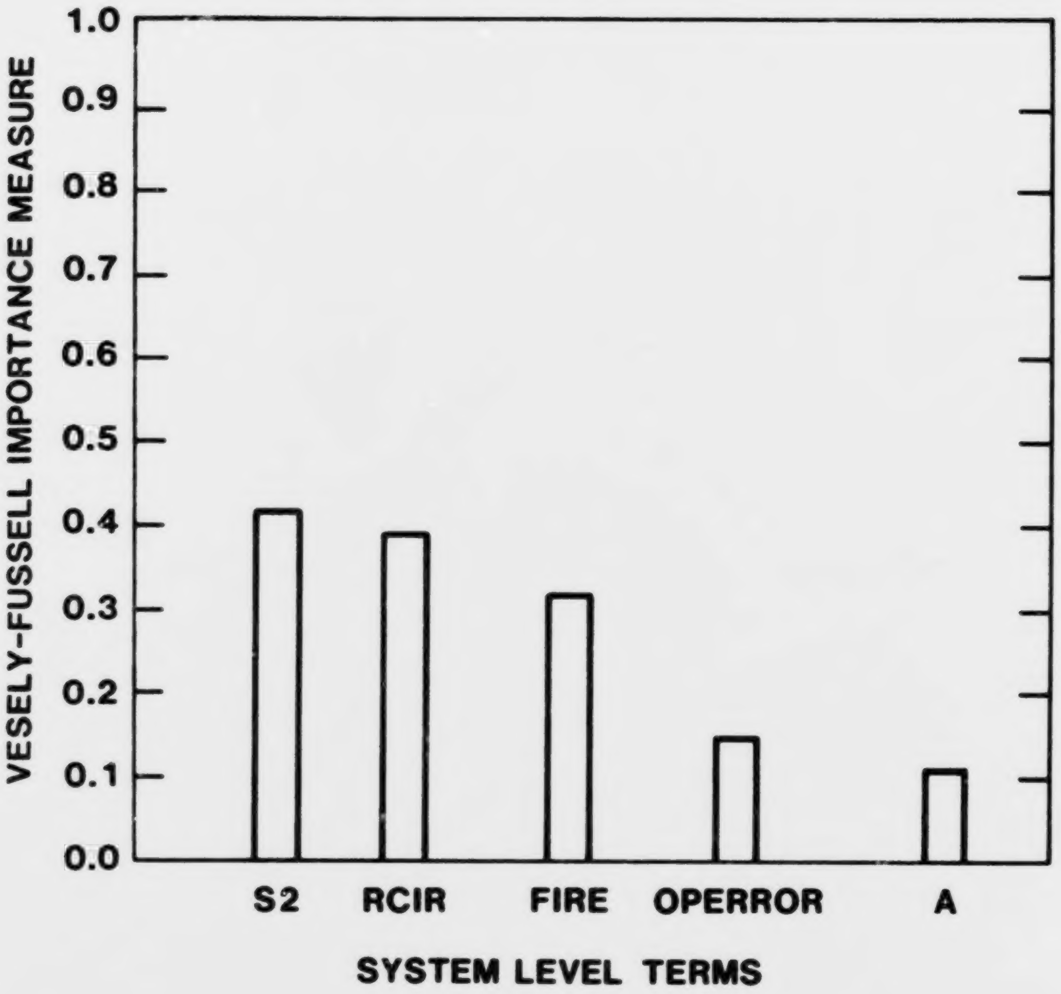

Figure 69. Indian Point 3, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criterion 2

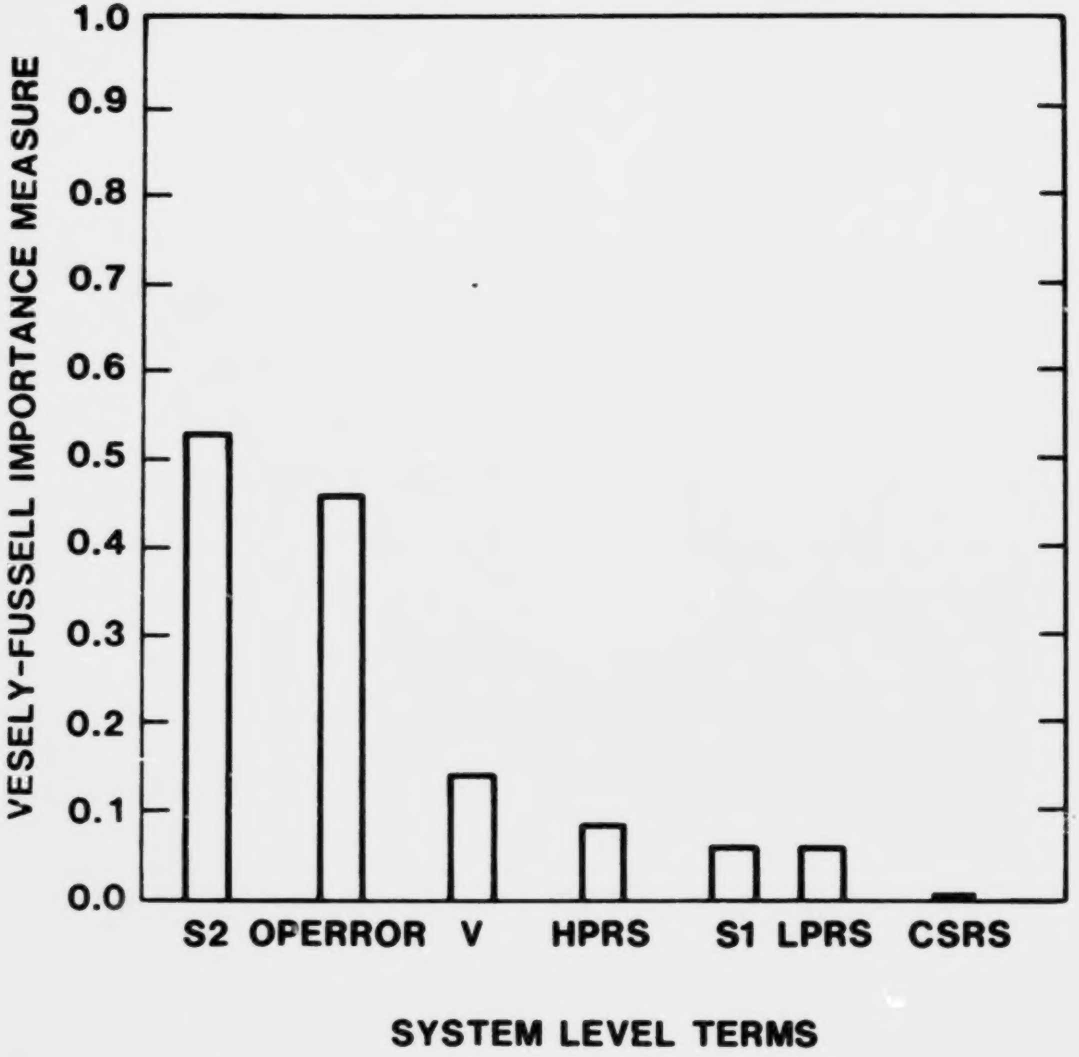

Figure 70. Sequoyah-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

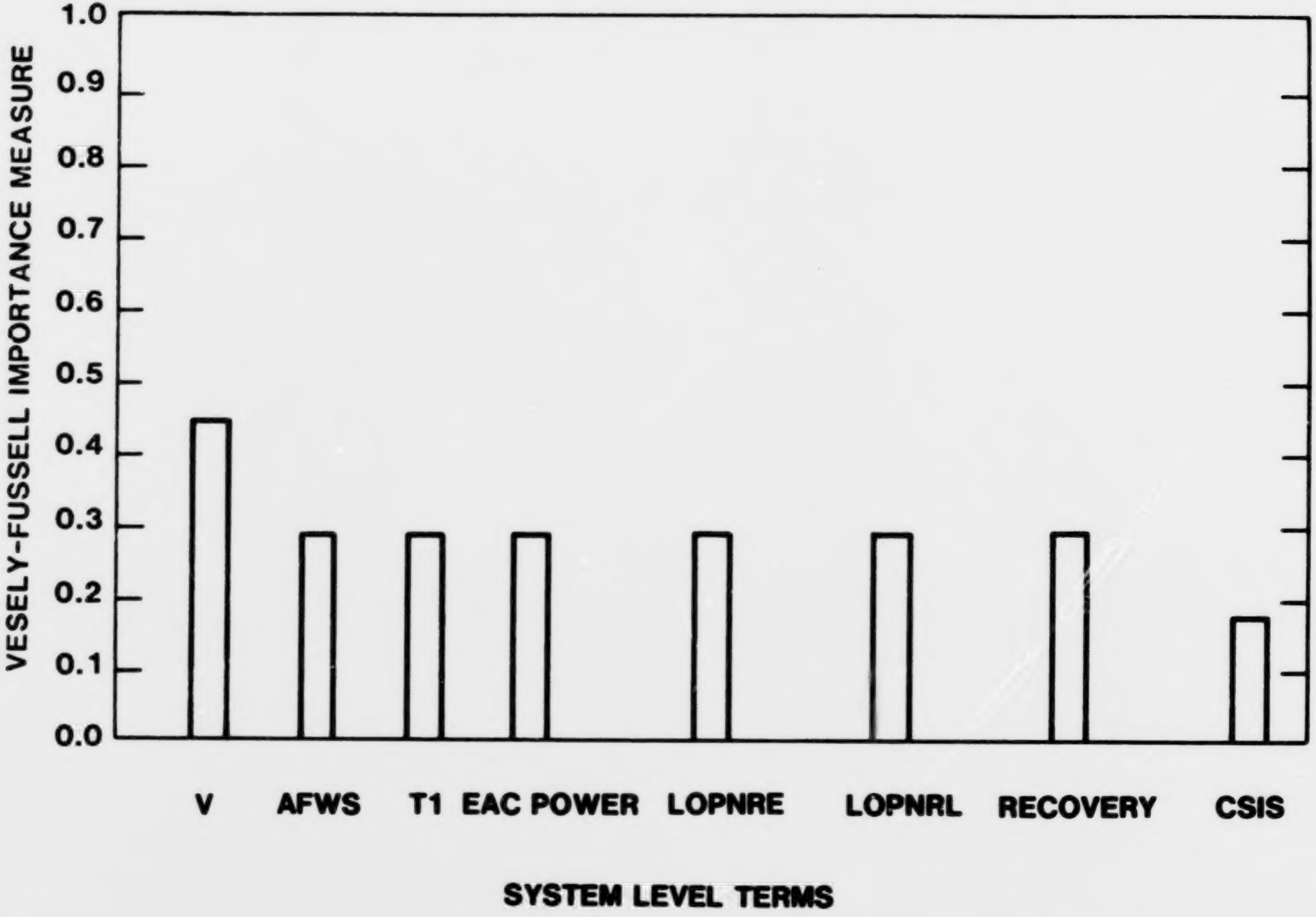

Figure 71. Surry, Importance of Front LIne and Support Systems to Total Plant Risk: Criterion 3

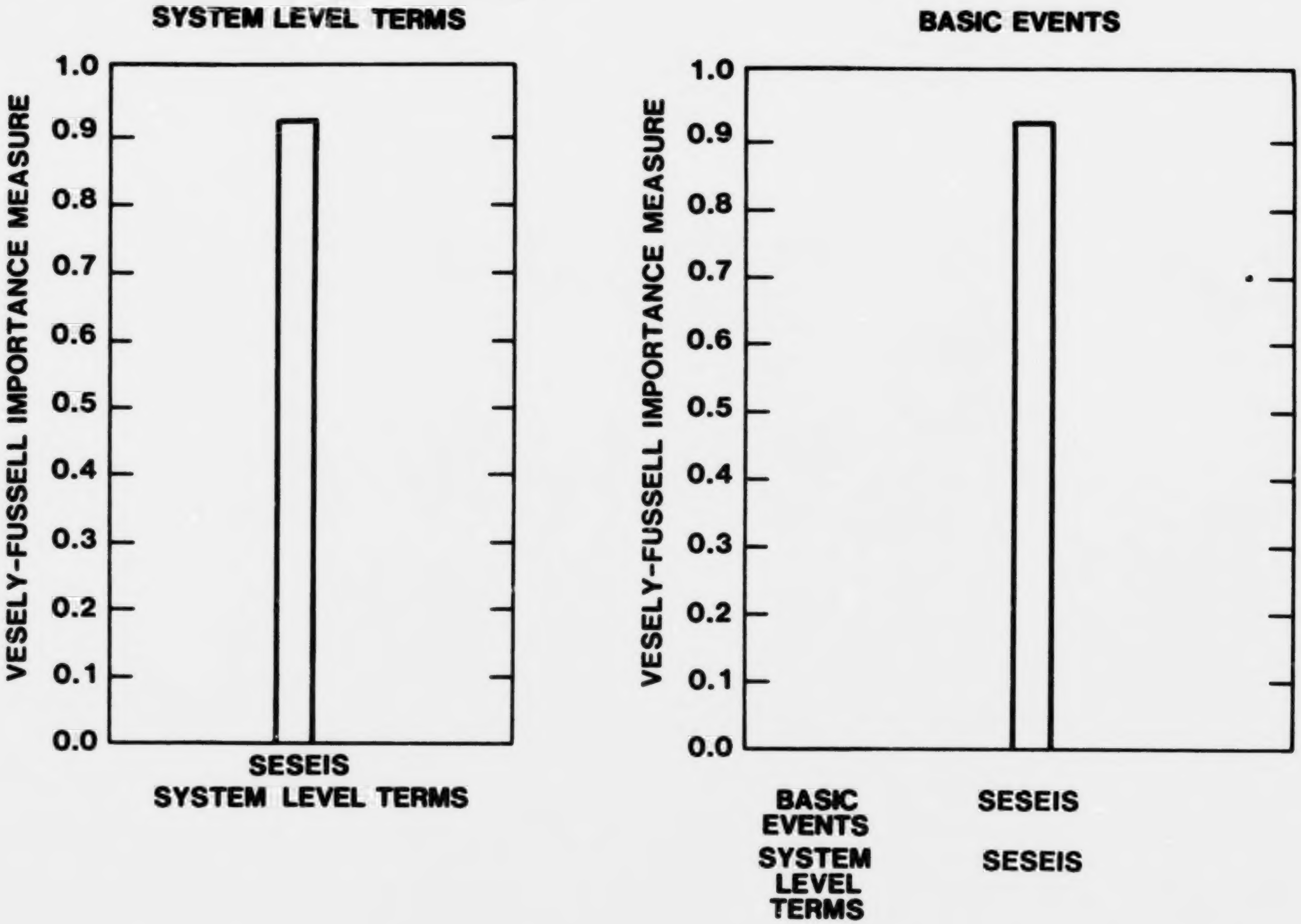

Figure 72. Zion, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

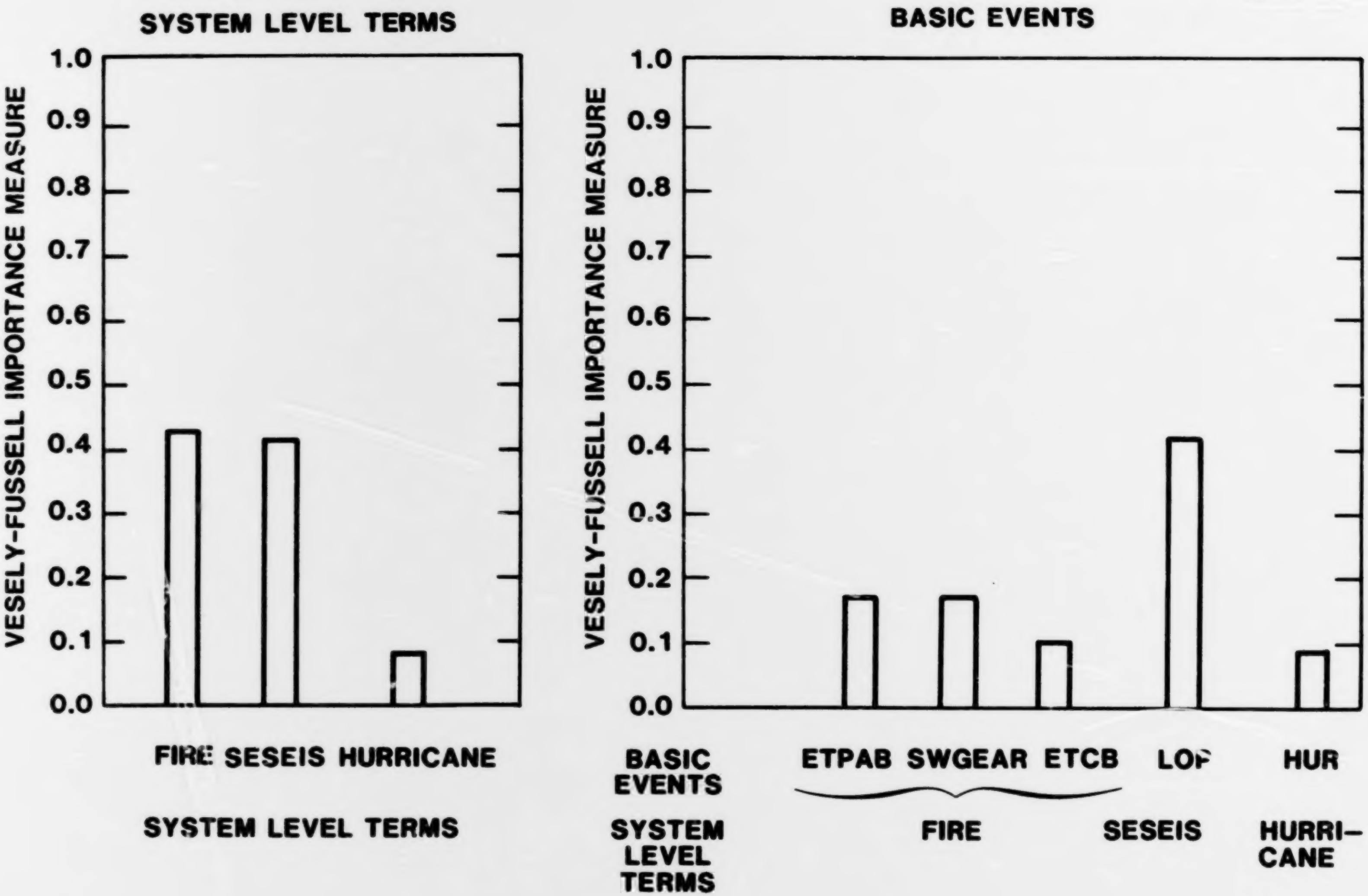

Figure 73. Indian Point 2, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

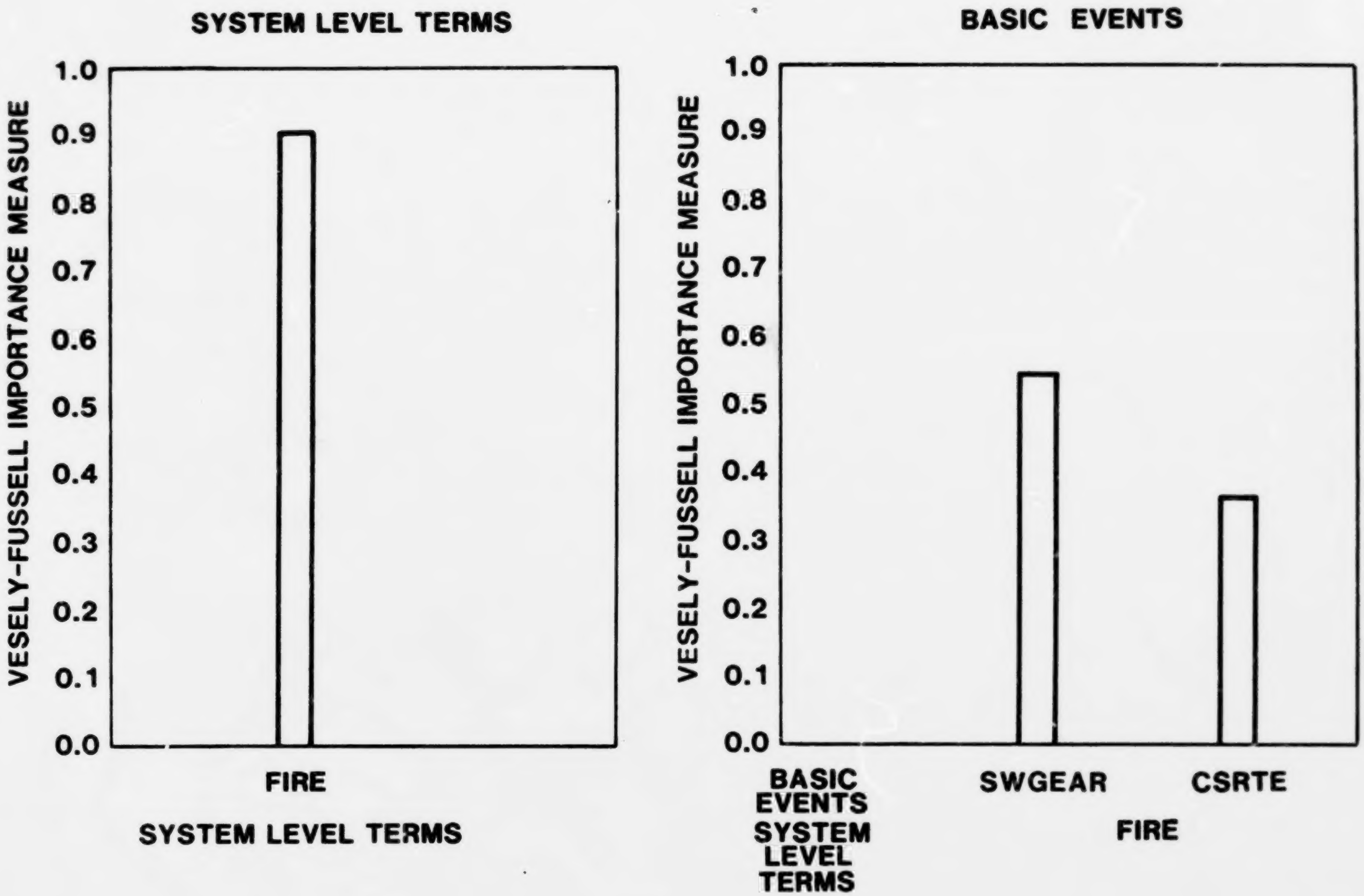

Figure 74. Indian Point 3, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

2. Babcock and Wilcox Plants

Crystal River-3
ANO-1
Oconee

a. Introduction

The PRAs performed on the above Babcock and Wilcox plants were very compatible to the importance methodology of this analysis. All of these PRAs modeled core melt sequences with sequence Boolean equations which are used to approximate the sequence probability equations. The dominant cut sets for each dominant sequence are published in the PRAs. It is these cut sets which are used for the importance calculation.

The three PRAs represent two different methodologies of PRA. The Oconee PRA is a Reactor Safety Study Methodology Applications Program (RSSMAP) Study, and the Crystal River and ANO PRAs are Interim Reliability Evaluation Program (IREP) studies. However, the presentation of sequence models (i.e., cut sets and basic events) in all three studies is similar. The level at which basic events are defined is basically the same in all three PRAs, although the Oconee Study defines each basic event in greater detail in the system appendices. The ANO study publishes two different values for such cut sets, one incorporating recovery, and one not incorporating recovery. The values incorporating recovery were used in this study.

The list below shows the percent of core melt frequency modeled by the cut sets published for the dominant sequences of each PRA:

Plant	Percent Core Melt Frequency
Crystal River	72%
ANO-1	66%
Oconee	68%

Approximately one-third of the core melt frequency is not modeled for each plants. About 5-10 percent of the unmodeled core melt frequency is due to eliminating all sequences contributing less than 5 percent to total core melt frequency from the analysis. The remaining modeling gap is due to the limitation on cut sets published in the PRAs for the dominant sequences. All three PRAs model about the same amount of each plant's total core melt frequency, so the results for each plant are based on the same quantity of information as for the other Babcock and Wilcox plants.

b. LOCA Initiating Events

Table 61 lists a summary of the various initiating events defined in the three PRAs of Babcock and Wilcox plants. The size limitations for LOCA events are those used for WASH-1400. The various LOCA events for the Babcock and Wilcox plants can be closely approximated into the WASH-1400 categories.

The Crystal River IREP Study defines plant specific LOCAs based on the Final Safety Analysis Report (FSAR) for Crystal River-3. The ANO and Oconee studies also developed plant specific data for LOCAs, using WASH-1400 methodology as a basis for calculating LOCA frequencies. Large and medium LOCA frequencies match very closely across the three plants. However, on the ANO model for small LOCAs is different than that for Crystal River and Oconee. The ANO study models Reactor Coolant Pump (RCP) seal ruptures as well as random pipe failures, where as the other two studies model only random pipe failures. This accounts for ANOs significantly higher small LOCA frequency. However, inspection of the dominant sequence frequencies for each plant (Crystal River - Table 48, ANO - Table 55, Oconee - Table 62) shows that the ANO small LOCA sequence frequencies are not necessarily greater than for Crystal River and Oconee.

The interfacing systems' LOCA, event V, is determined to be significant only at Oconee. The Oconee PRA models this event similarly to WASH-1400. The ANO and Crystal River studies claim that testing procedures and information feedback to the operators in the event of a check valve failure are sufficient to reduce the probability of interfacing system LOCAs below significant levels. It is not clear whether the event V frequency at Oconee is due to assumptions in the PRA methodology or to plant specific features. The Oconee PRA makes no mention of Oconee plant testing procedures, or lack thereof, affecting interfacing systems LOCAs. Nor does the study mention operator recognition of such a LOCA. Event V is a significant event only to plant risk at Oconee. Small LOCA sequences are dominant to risk and core melt frequency at all three plants. At Oconee, one large LOCA sequence, S_1D, also is dominant to core melt frequency.

c. Transient Events

Oconee transient events and frequencies were taken from WASH-1400. The ANO and Crystal River studies used data from EPRI NP-801[22] to develop transient frequencies. In addition, the ANO study considers failures of normally operating systems as transient events. While the ANO transient events are more detailed in their grouping than for Oconee and Crystal River, the main grouping of transient events and their frequencies are reasonably similar at the three plants.

d. External Events

No external events were considered for the ANO, Crystal River, and Oconee studies.

e. Containment Response

The containment response models for the ANO and Crystal River studies are based on the Oconee containment response model. Sequences for ANO and Crystal River were compared to similar sequences at Oconee to assign containment failure modes and release category probabilities to ANO and Crystal River sequences. Certain plant specific considerations were used in the ANO study for some sequences, but no significant difference exists between the containment response models for the three plants.

f. Dominant Systems and Basic Events

The system level importance results for Babcock and Wilcox plants (Figures 74 through 80) show that operator errors clearly dominate at Crystal River and Oconee. Operator errors at ANO are also important, but not significantly more or less important than other dominant system level failures at ANO.

Human failures at Crystal River and Oconee involve improper operator actions during switchover to high pressure recirculation, or improper actions during recirculation, such as shutting down low pressure pumps too soon. It is not surprising that these types of human errors dominate at these two plants. The two most dominant sequences for each plant involve HPRS related operator error. These are the Oconee sequences T_2MQ-H and S_3H, and the Crystal River sequences B_4S_2 and B_4S_{23}. Sequence B_4S_{23} actually involves a failure of emergency injection, but the operator error causing this failure is premature switchover to HPRS. Operator failures related to HPRS switchover and operation at Crystal River are much more important than at Oconee. The Crystal River study uses probabilities of operator failures of approximately an order of magnitude higher than the Oconee and ANO studies. Frequencies of sequences involving recirculation failures at ANO are not significantly smaller than the ANO dominant sequence frequencies listed on Table 55. But these sequences are eliminated by the limitation in this analysis of studying only sequences contributing 5 percent or more to core melt frequency. Operator errors at ANO are dominated by the operator failure to initiate HPIS after an ATWS transients, and after a very small LOCA (B1.2) where it is assumed the automatic actuation pressure of HPIS is never reached. The ATWS sequence involves a high stress environment for the operators, and operator error is assigned a significantly higher frequency for the ATWS sequence than for the LOCA sequence.

A significant difference of system level importances between the three plants is DC power system importance. The Oconee study assigns an insignificant value of DC system unavailability. Thus, there are no DC terms in the cut sets for Oconee. The unavailability of Oconee's DC power system is discussed in Appendix B2 of the Oconee study. AT ANO, DC power is the single most important system level term. DC power unavailability is dominated by a common mode failure of both station batteries, which fails all mitigating systems (EBAT-CM). The importance of DC power at Crystal River is dominated by failure of the B battery, which fails both the B diesel and the turbine driven emergency feedwater pump. The unavailabilities of the total DC systems at ANO and Crystal River are approximately the same, 1.0E-5. However, for ANO, failure of both batteries is in a two event cut set of sequence $T(LOP)2D_1YC$. The value of this one cut set is 8.3E-6, which by itself is greater than the frequency of any other dominant sequence at ANO. At Crystal River-3, failure of the B battery is backed up by the A diesel and backup AC power from Crystal River Units 1 and 2, and appears in four term cut sets of sequence T_2AT_{10}. Furthermore, the importance of DC power at Crystal River is diminished by the dominance of sequence B_4S_2, and its two event cut sets involving human errors.

The results for the Babcock and Wilcox plans for Criterion 3 (Figures 24, 28, 32) are essentially the same, although certain differences are noted. Event V is significant to risk at Oconee, as discussed above in Section III.C.2. The Oconee sequence involving LPIS failure, S_1D, is not dominant to risk, due to its release category probabilities (Table 62). Thus, LPIS is not significant to risk at Oconee, where it is significant to core melt frequency. DC power at ANO is significant to risk as well as core melt frequency. However, DC power's importance to risk is only one-third of its importance to core melt frequency. This is due to the release category probabilities of sequence $T(LOP)\ LD_1YC$ (Table 55), which weight the sequence cut sets down with respect to risk compared to the other sequences.

g. Summary

The results for the three Babcock and Wilcox plants indicate that plant specific designs can significantly impact which systems are important to core melt accidents. The difference in importance of DC power at the various plants is the most obvious example of how plant specific design affects systems importance. Event V is also a good example of this. The importance of operator actions varies significantly between the three plants. It is not clear from the PRAs whether this is due to plant specific situations which impact operator actions, or to modeling differences in the PRAs.

Table 120

Frequency of Initiating Events $(R\text{-}yr.)^{-1}$
Babcock and Wilcox Plants

	Crystal River 3		ANO-1		Oconee	
Initiating Event	Initiating Event	Frequency	Initiating Event	Frequency	Initiating Event	Frequency
1. Large LOCA < 6"	B1	1.0E-4	B(10)	1.6E-4	A	1.0E-4
			B(13.5)	1.2E-5	S_1	1.0E-4
			B(> 13.5)	7.5E-5		
2. Medium LOCA 2" < 6"	B_2	1.0E-4	B(4)	3.8E-4	S_2	4.0E-4
3. Small LOCA < 2"	B_3	1.0E-4	B(1.66)	3.1E-4	S^3	1.3E-3
	B_4	1.3E-3	B(1.2)	2.0E-2		
4. Interfacing Systems LOCA	V	2.0E-9	-	-	V	4.E-6
5. Transient Loss of Offsite Pwr	T_{2A}	0.32	T(LOP)	.32	T_1	.2
6. Transient Loss of Main Feedwater or PCS	$T_{2A}-T_{2A}$	2.0	T(PCS)	1.0	T_2	3.0
7. Other Transients	T_1	8.5	T(FIA)	7.1	T_3	4.0
			T(A3)	3.5E-2		
			T(B5)	3.5E-2		
			T(D01)	1.8E-2		
			T(D02)	1.8E-2		
			T(LOSW)	2.6E-3		

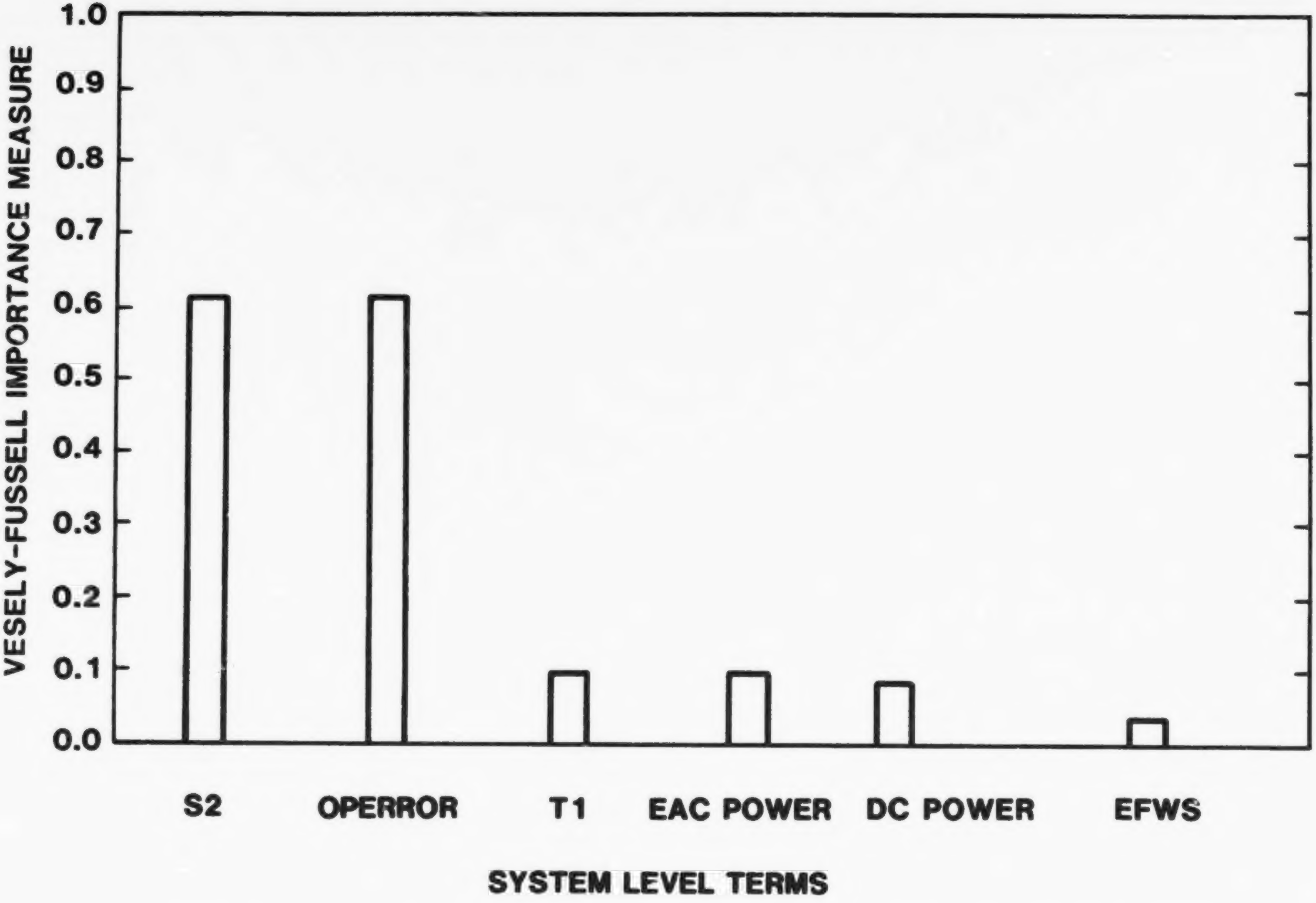

Figure 75. Crystal River 3, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

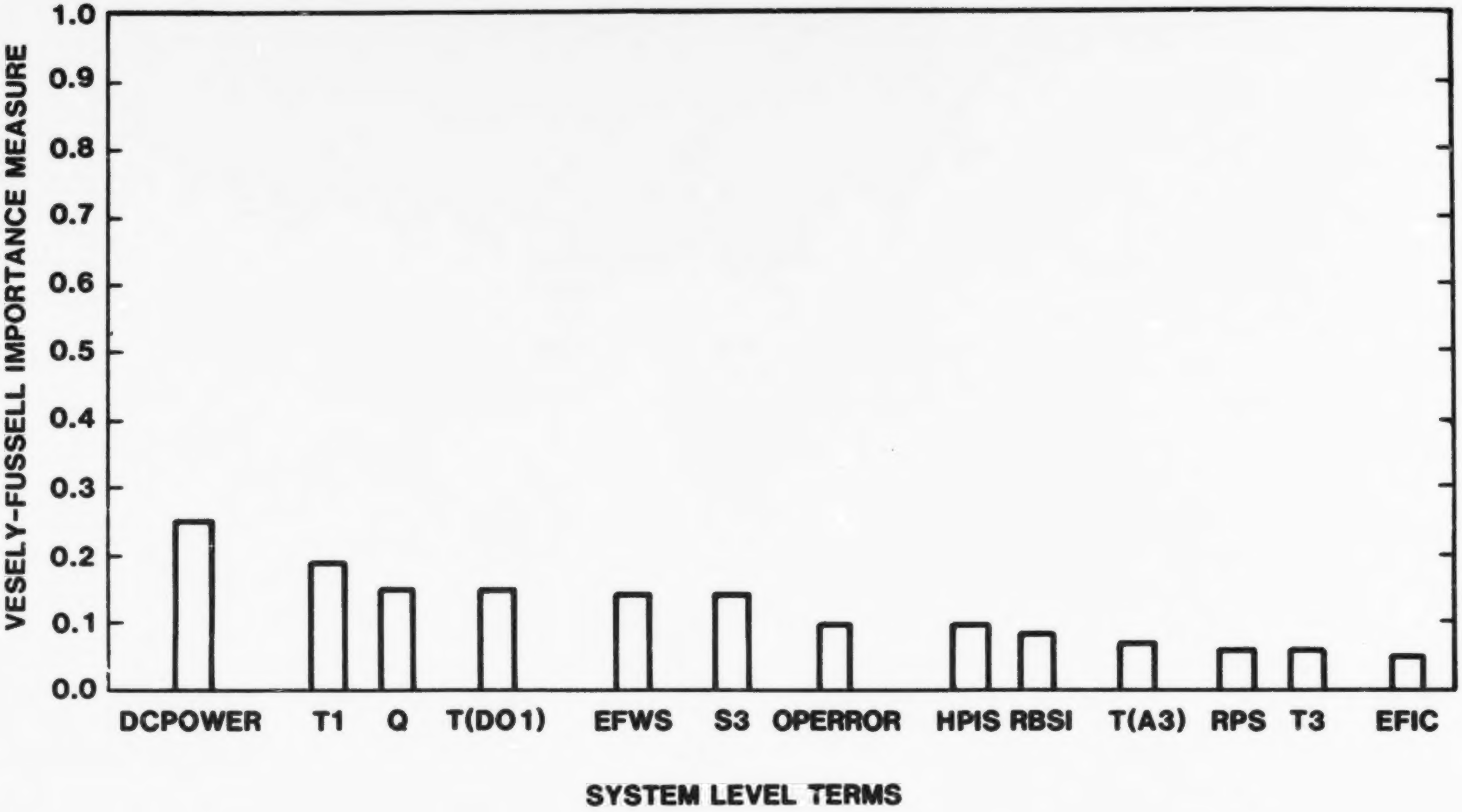

Figure 76. ANO-1, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

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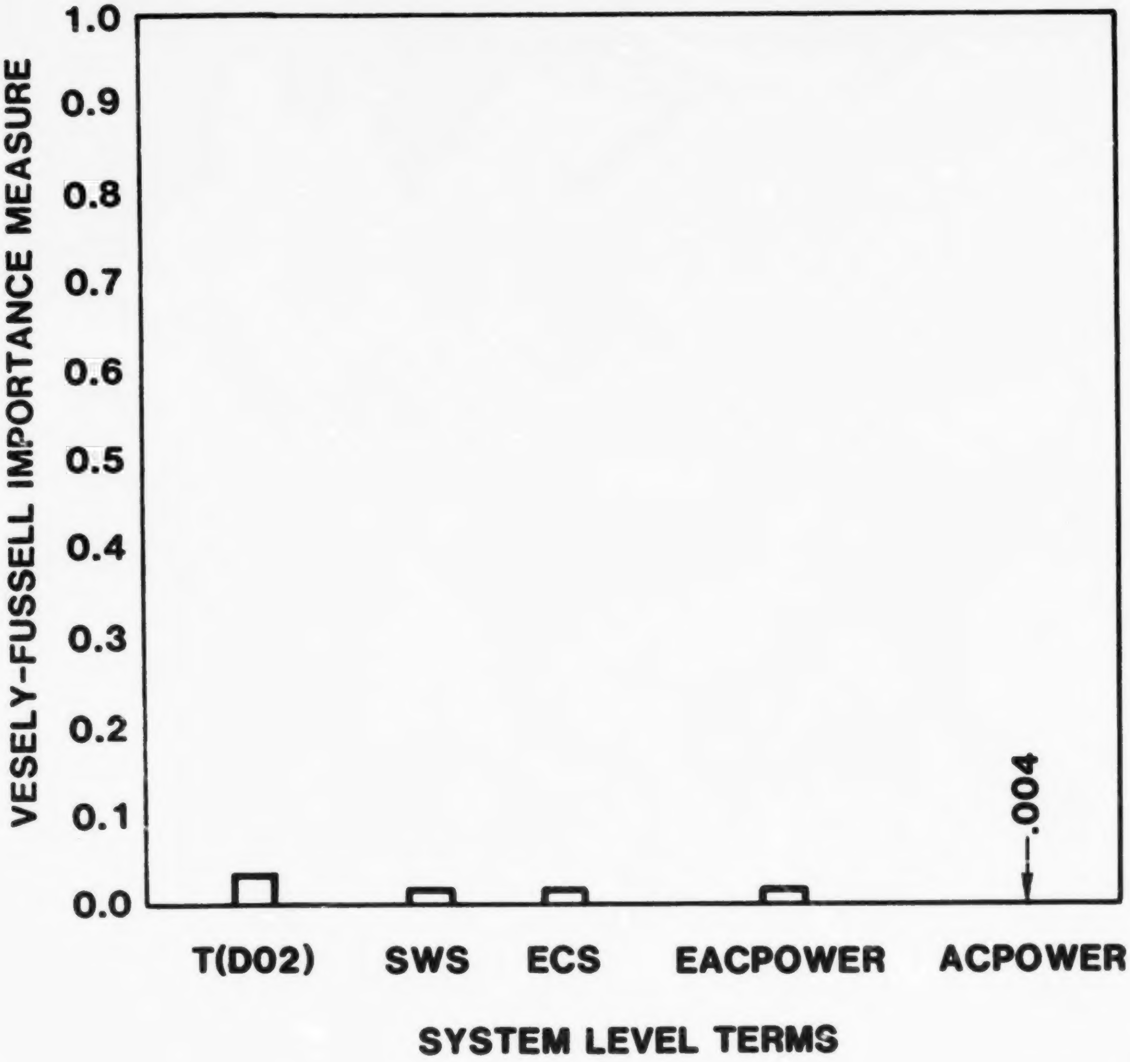

Figure 76. ANO-1, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2 (Cont)

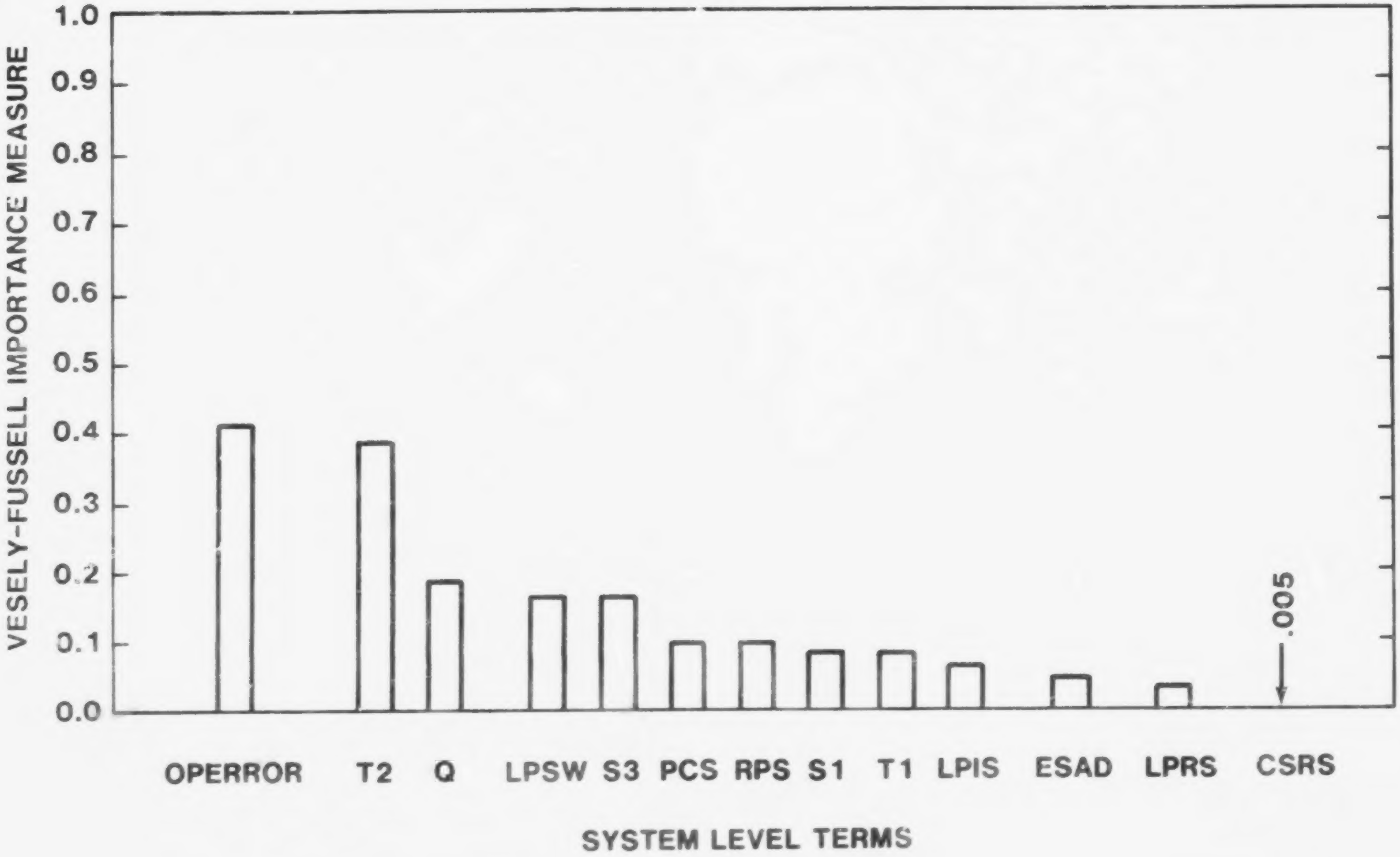

Figure 77. Oconee, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

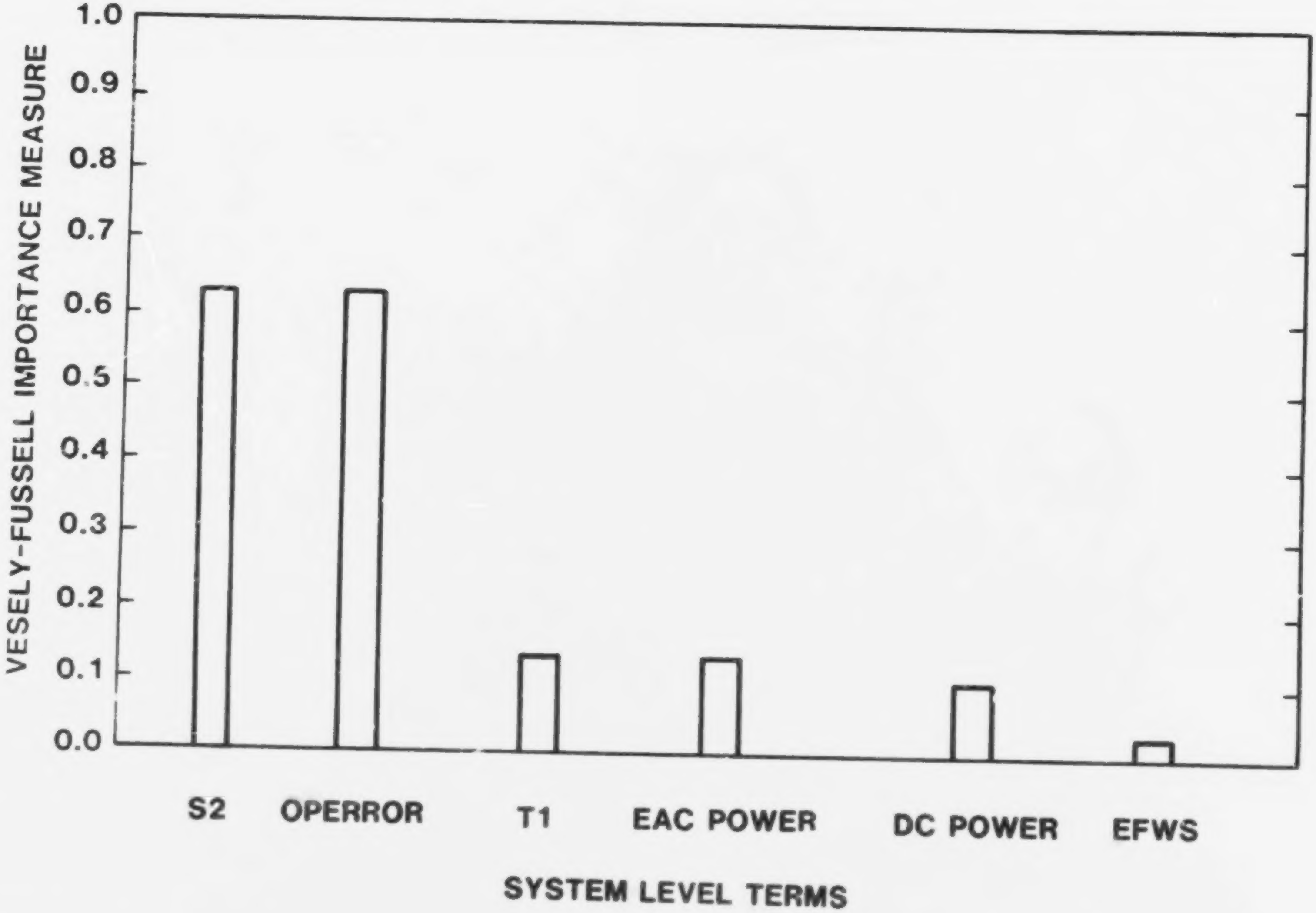

Figure 78. Crystal River 3, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

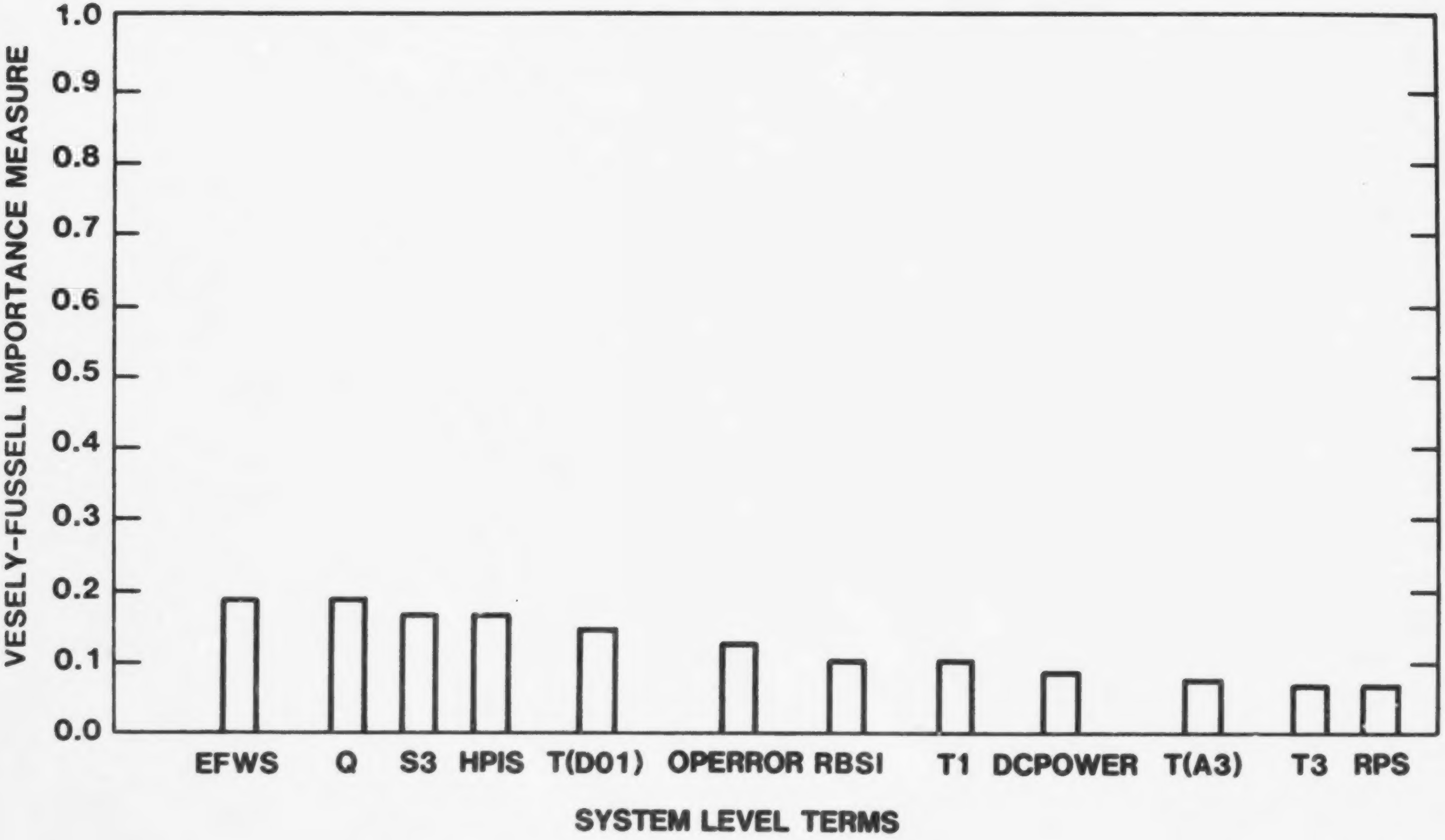

Figure 79. ANO-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

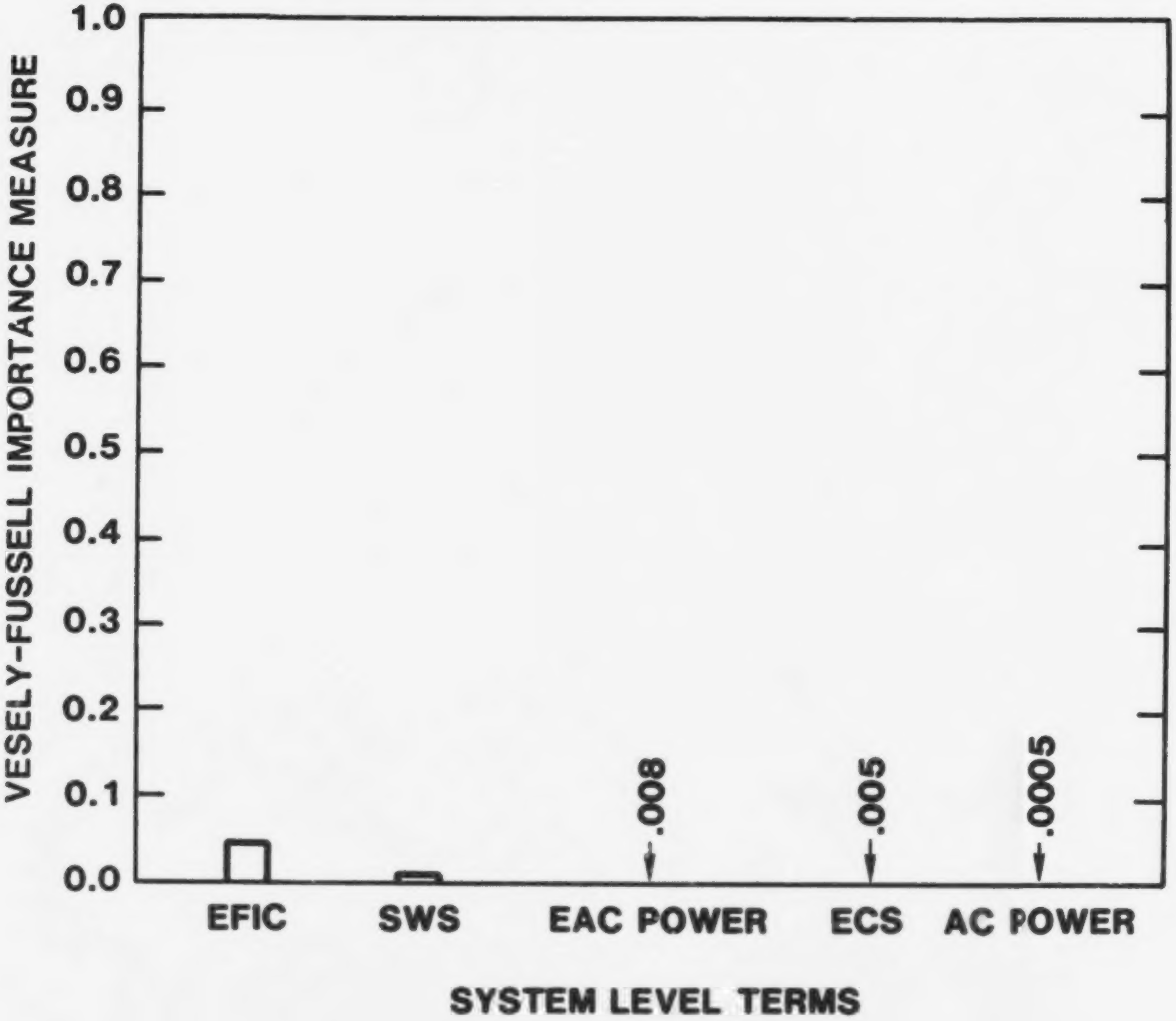

Figure 79. ANO-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3 (Cont)

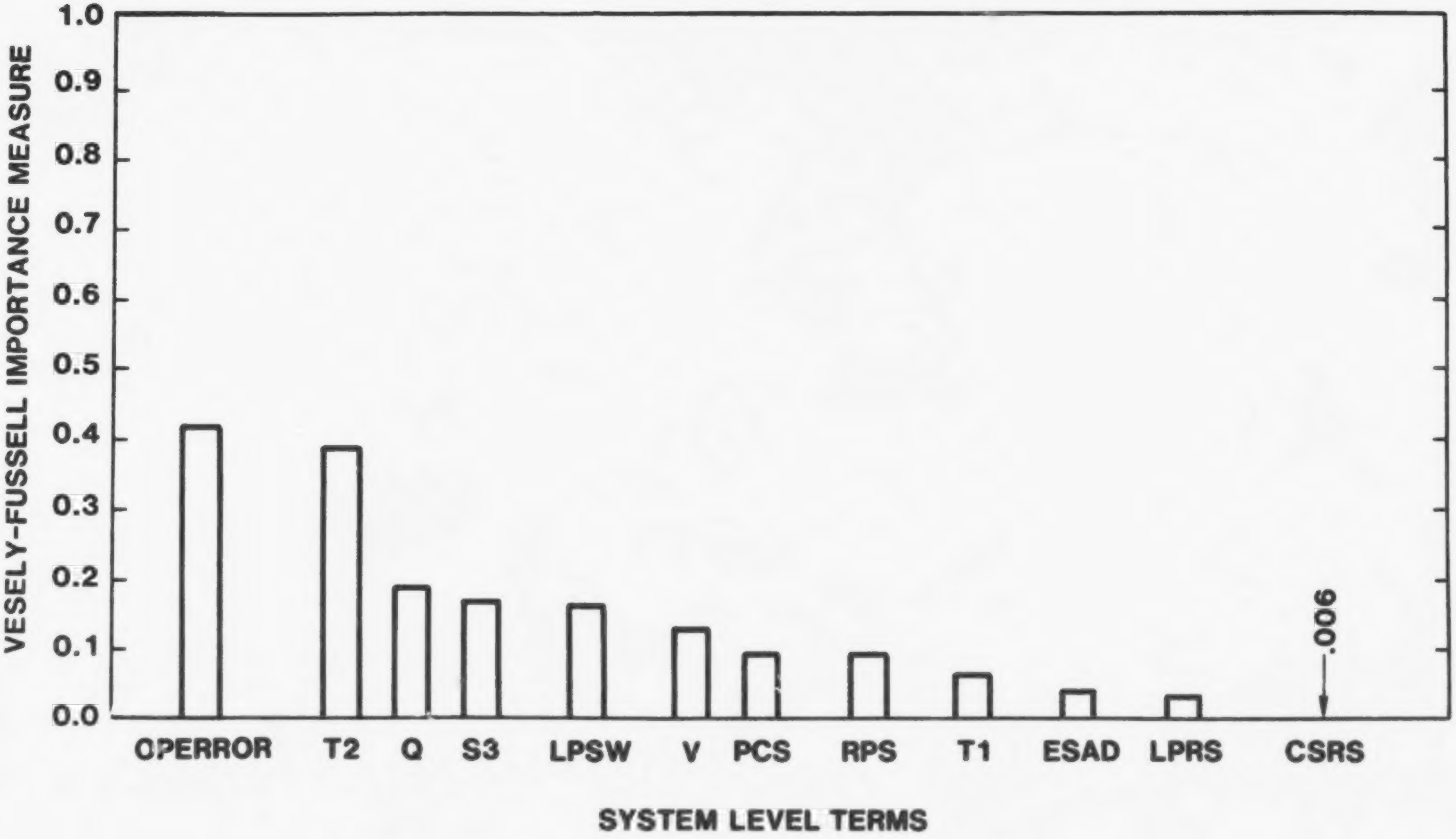

Figure 80. Oconee, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

3. Combustion Engineering Plants

a. Calvert Cliffs

Only one PRA has been published on a Combustion Engineering plant, the Calvert Cliffs Reactor Safety Study Methodology Application Program (RSSMAP) study. So, no inter-plant comparisons can be made for Combustion Engineering plants.

4. General Electric Plants

Browns Ferry-1
Grand Gulf
Peach Bottom (WASH-1400 PWR)
Limerick
Big Rock Point
Millstone

a. Introduction

Of the five PRAs performed on GE plants, only the Grand Gulf Reactor Safety Study Methodology Application Program (RSSMAP) study is highly compatible to the importance methodology developed for this study. The Grand Gulf study presents sequence Boolean equations for the dominant sequences and lists the dominant cutsets for each sequence. The Boolean equations and the cutsets of the equations are used as approximations of probabilistic equations for the importance calculations.

The Browns Ferry-1 PRA is an Interim Reliability Evaluation Program (IREP) study. However, unlike the two other IREP studies analyzed for this report, Crystal River and ANO, the Browns Ferry study does not publish sequence Boolean equations and dominant cutsets for the dominant sequences. Sequence cutsets were generated for Browns Ferry using the system cutsets in Appendix B and the sequence logic diagrams in Appendix C of the study. However, due to the limited information supplied in the study on system cutsets, only 40 percent of the total core melt frequency at Browns Ferry could be modeled with the resulting cut sets. The lack of sequence modeling for Browns Ferry poses serious questions as to the utility of our Browns Ferry results.

The remaining three PRAs done on GE plants are the WASH-1400 PWR (Peach Bottom), and two independent studies, the Limerick PRA and the Big Rock Point PRA. None of these studies provide sequence Boolean equations and dominant cutsets for the dominant sequences. The Peach Bottom study does provide somewhat detailed descriptions and quantification of system failures. However, the cutsets developed from the simplified fault trees constructed for

Peach Bottom for this report are very general in nature. These system level cutsets do not provide for a satisfactory application of the importance methodology, since the basic events for Peach Bottom are very general and common failures between basic events cannot be taken out of the resulting Boolean equations. The same problem applies for the Limerick and Big Rock Point Studies. However, the Limerick and Big Rock Point Studies provided no detailed description or quantification of system failures or event unavailabilities. The basic event for Limerick are at the event tree level, and the Big Rock Point basic events are taken from every general sequence descriptions.

The list below shows the percent of core melt frequency modeled by the cut sets for each plant:

Plant	Percent Core Melt Frequency
Browns Ferry	40%
Grand Gulf	82%
Peach Bottom	87%
Limerick	70%
Big Rock Point	94%
Millstone	50%

The percent of core melt frequency modeled for Browns Ferry clearly casts a dubious light upon the results for that study. While the cut sets for the Peach Bottom and Big Rock Point studies model significant portions of the core melt accidents, the general nature of these cut sets places significant limitations on the utility of the results for these plants. The Limerick cut sets are not only general in nature, but fail to model 30 percent of the core melt frequency. The Grand Gulf RSSMAP study is the only PRA published on a BWR plant which makes its core melt accident models sufficiently available for further analyses, such as the importance calculations of this report. The inter-plant results discussed below should be studied with the significant limitations of the various PRAs kept in mind.

b. LOCA Initiating Events

Table 121 summarizes the initiating events developed for the various PRAs on GE plants. The size limits for large, medium, and small LOCAs are those used in WASH-1400 for Peach Bottom. The Limerick, Big Rock Point, and Browns Ferry LOCAs approximate quite well to these limits. The Grand Gulf study team decided to split all LOCAs into two categories, LOCA diameters greater than 13.5 in, and LOCA diameters less than 13.5 in.

The Grand Gulf and Browns Ferry PRAs used WASH-1400 methodology with plant specific data to calculate LOCA frequencies. The LOCA frequencies are very similar between the three studies. Limerick and Big Rock Point studies used specific methodologies to identify and quantify LOCAs. The Limerick LOCA frequencies for medium and small LOCAs are an order of magnitude higher than for the other plants. No discussion or reason for these values is given in the Limerick study. The Big Rock Point LOCA frequencies compare well with the WASH-1400 values.

The interfacing LOCA event V was found to be insignificant for all the BWR plants except Big Rock Point. However, no discussion of the nature of an interfacing LOCA could be found in the BRP study.

c. Transient Events

The transient initiating events for each plant are also summarized on Table 121. Peach Bottom and Grand Gulf values are slightly different from each other, reflecting differences in plant experiences. Browns Ferry transients were identified and quantified using EPRI NP-801.[22] The Limerick and BRP studies use their own methodologies and selected NRC, EPRI, and GE documents to identify and quantify transients. The most significant difference between the plants is for transients with loss of offsite power. BRP, Grand Gulf, and Peach Bottom have similar values. But Limerick and Browns Ferry are an order of magnitude lower.

d. External Events

Only the BRP considered external events as accident initiators. Thus, comparisons of external events between the various GE plants is impossible.

e. Containment Response

The Browns Ferry, Grand Gulf, and Peach Bottom studies all use WASH-1400 BWR release categories to model accident sequences. The Browns Ferry study bases many of its containment response conclusions on the WASH-1400 model, plus certain plant specific information. Both Peach Bottom and Browns Ferry are Mark I containments. The Grand Gulf containment response analysis was performed using the MARCH and CORRAL computer codes, as was WASH-1400. However, unlike the Mark I containment, the Mark III containment atmosphere at Grand Gulf is not inerted. So, the Grand Gulf analysis had to consider hydrogen burning.

The Limerick study's containment response analysis splits sequences into its own containment failure modes, which where correlated to WASH-1400 BWR release categories, as described in Section III.B.1 under the Limerick results.

The BRP study modeled its containment response using the same computer codes as WASH-1400, plus the INCOR code developed by SAI and the INTER code developed by Sandia National Laboratories. The BRP study defined its own release categories and consequences for these categories. Like WASH-1400, there are five release categories for BRP, with category 1 the most dangerous with respect to impact on the public, and category 5 the least dangerous.

f. Dominant Systems and Basic Events

The bar graphs illustrating the system importance results for the GE plants in Section III.B.1 are included again here as Figures 81 through 86 for core melt frequency and Figures 87 through 92 for plant risk.

A cursory review of these results indicate that system importances vary significantly from plant to plant. For example, operator error is significant only at Peach Bottom and BRP; the Reactor Protection System (RPS) is important at Peach Bottom, Grand Gulf, and Browns Ferry; Injection Systems are very significant at Limerick and Peach Bottom only; emergency AC power is significant only at Grand Gulf and Browns Ferry. It is not surprising that no major correlation of system importances exist between the five GE plants. The results of the other PRAs analyzed in this report have shown system and component importances to be very dependent on the specific plants in question. Comparisons between the five GE plants is further complicated by the nature of the sequence models for Limerick, Peach Bottom, and BRP. The degree of information regarding sequence basic events provided in these three PRAs permitted little detail in the sequence models. Meaningful calculation of component importances are virtually impossible, and the system level importances are questionable. Furthermore, the Browns Ferry sequence models developed for this analysis are so limited by the information published in the PRA that the utility of the results for Browns Ferry is minimal.

Table 121

Frequency of Initiating Events $(R\text{-}yr)^{-1}$
General Electric Plants (BWRs)

	Browns Ferry-1		Grand Gulf		Peach Bottom	
Initiating Event	Initiating Event	Frequency	Initiating Event	Frequency	Initiating Event	Frequency
1. Large LOCA < 6"	D V	5.0E-5 5.2E-5	A S	1.0E-4 1.4E-3	A	1.0E-4
2. Medium LOCA 2" < 6"	I_v I_v	9.0E-5 2.1E-4	S	-	S_1	3.0E-4
3. Small LOCA < "2	S	1.0E-3	S	-	S_2	1.0E-3
4. Interfacing Systems LOCA	-	-	V	0.0	V	0.0
5. Transient Loss of Offsite Power	T_p T_{pk}	3.0E-2 1.7E-3	T_1	.21	TxRLOP	0.2
6. Transients Not Involving Loss of Offsite Power	T_A T_k T_u	1.7 .16 1.7	T_{23}	7.0	TxRLOP/	9.8
7. External Events	-	-	-	-	-	-

Table 121 (Cont.)

Frequency of Initiating Events $(R\text{-}yr)^{-1}$
General Electric Plants (BWRs)

	Limerick		Big Rock Point	
Initiating Event	Initiating Event	Frequency	Initiating Event	Frequency
1. Large LOCA < 6"		4.0E-4		1.0E-5
2. Medium LOCA 2" < 4"		2.0E-3		
Medium Steam Line Break Inside Containment			S_4	1.0E-4
3. Small LOCA < "2		1.0E-2		1.0E-3
4. Interfacing Systems LOCA	-	-	IL	2.0E-3
5. Transient Loss of Offsite Power	T_E	5.3E-2	LOSP	.13
6. Transients Not Involving Loss of Offsite Power	T_T	4.0	Spurious Bypass Valve, B	.01
	T_P	2.0	Loss of Instrument Air, U	.06
	T_I	0.7		
	Other Transients	1.0	Other Transients	3.2
7. External Events	-	-	Fire (D_P)	1.8E-3

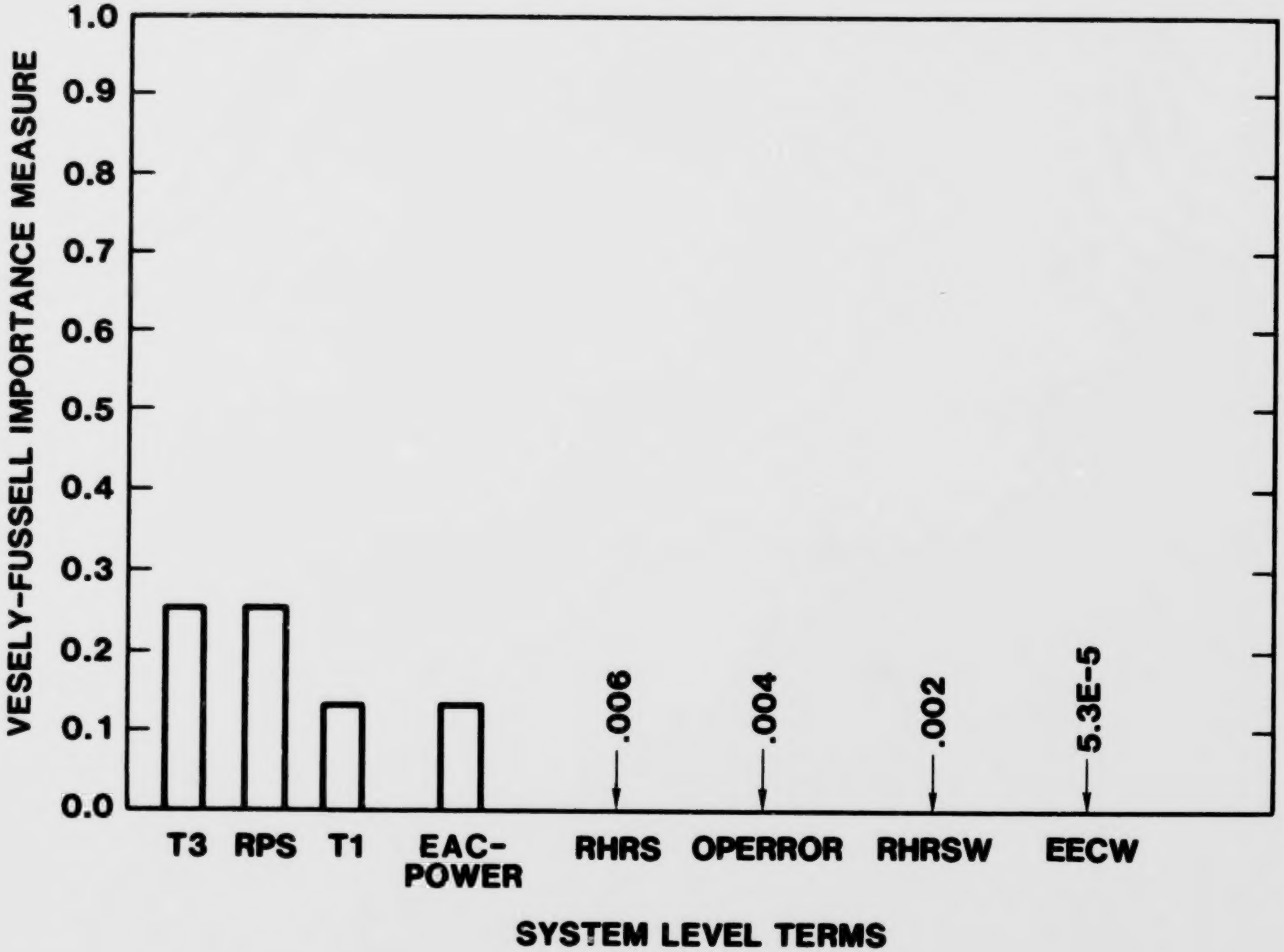

Figure 81. Browns Ferry-1, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

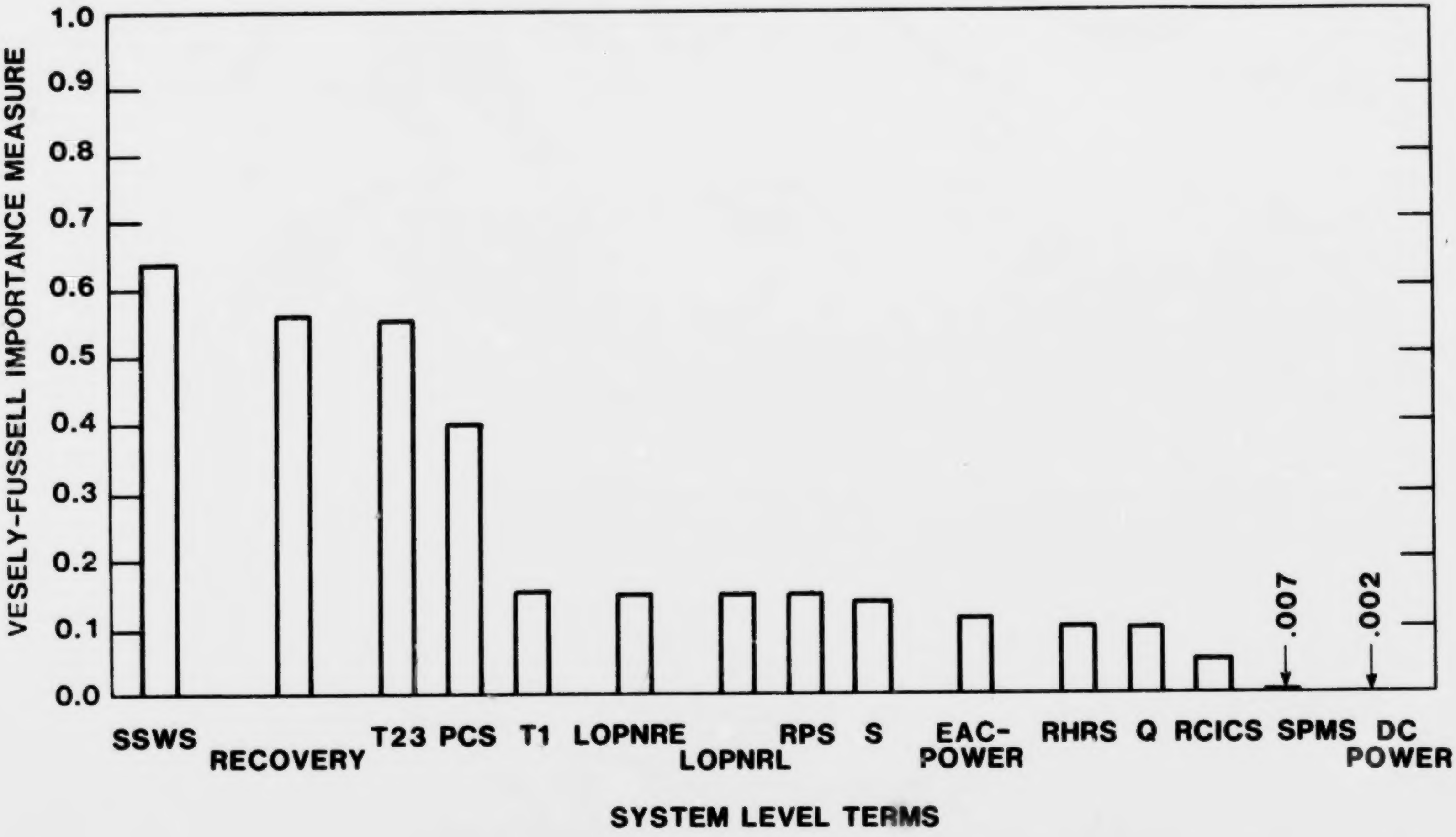

Figure 82. Grand Gulf, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

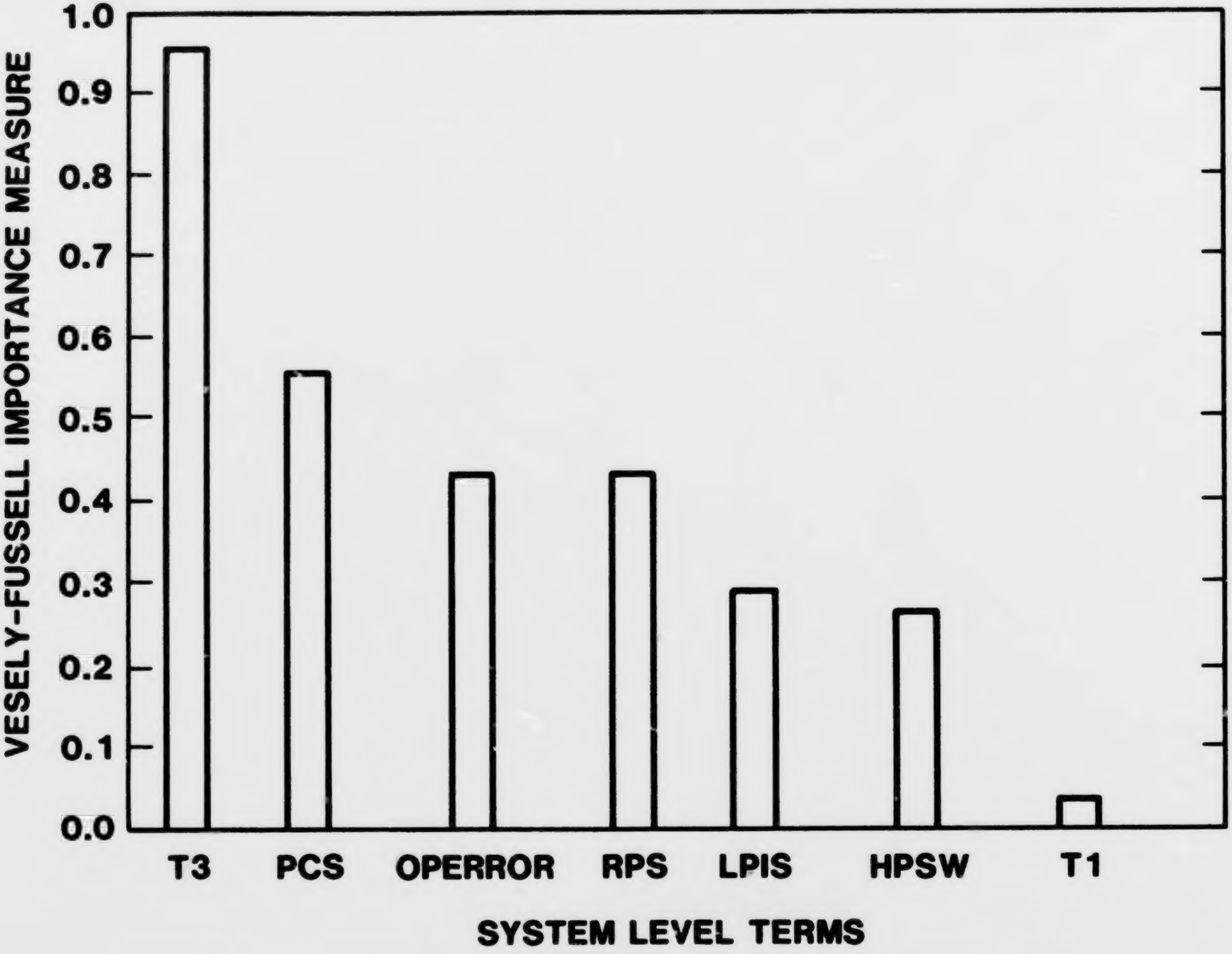

Figure 83. Peach Bottom, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

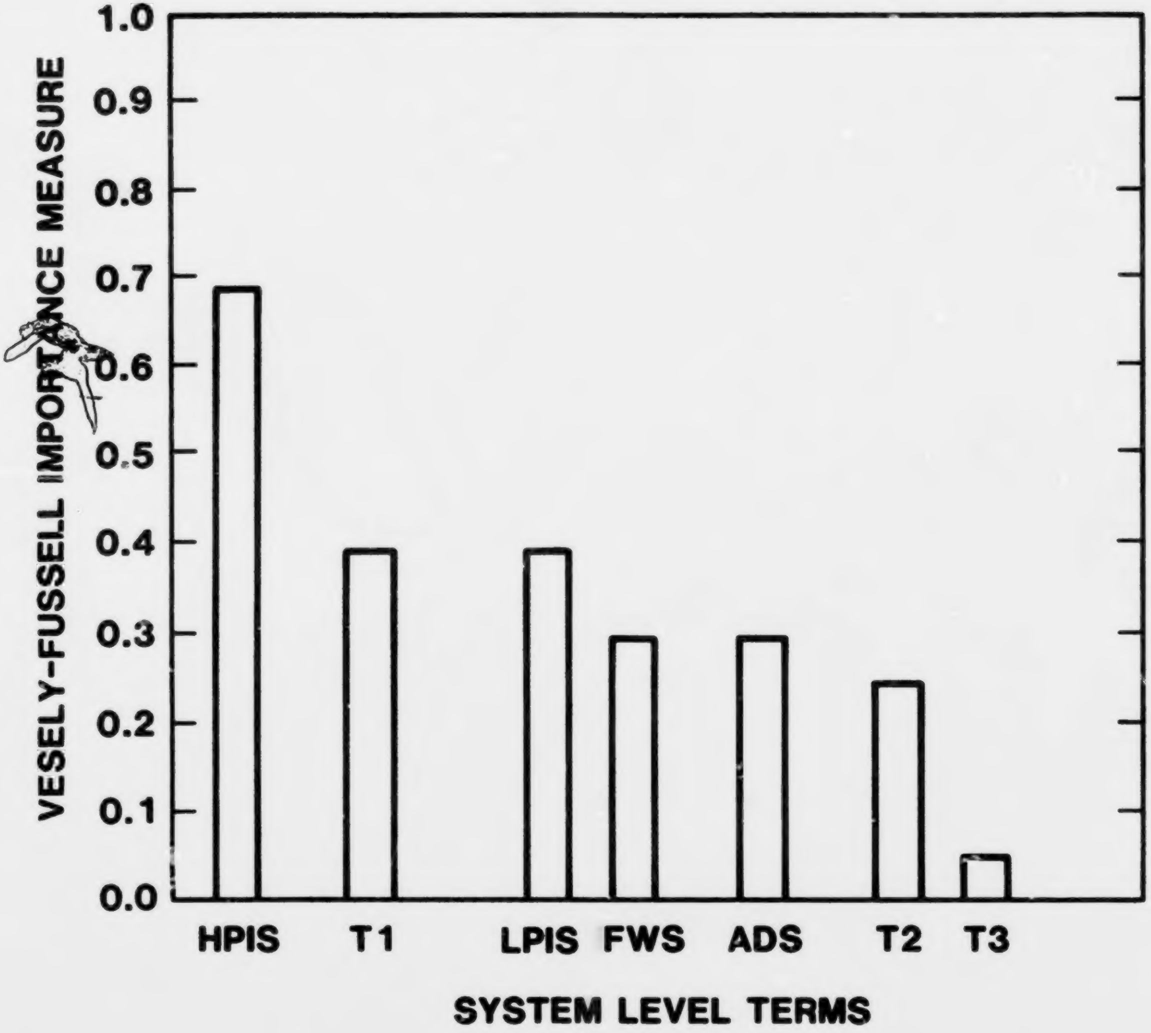

Figure 84. Limerick, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

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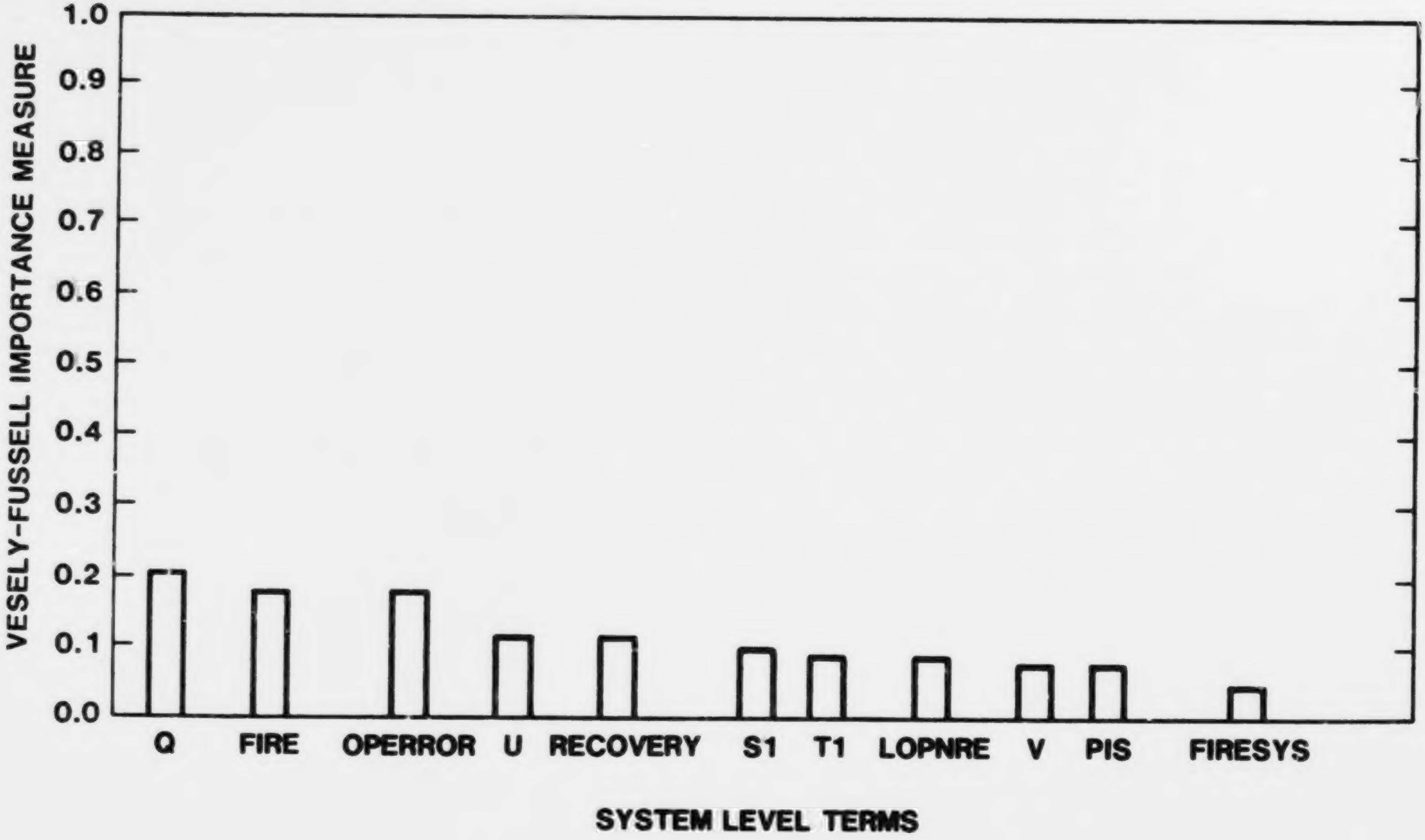

Figure 85. Big Rock Point, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

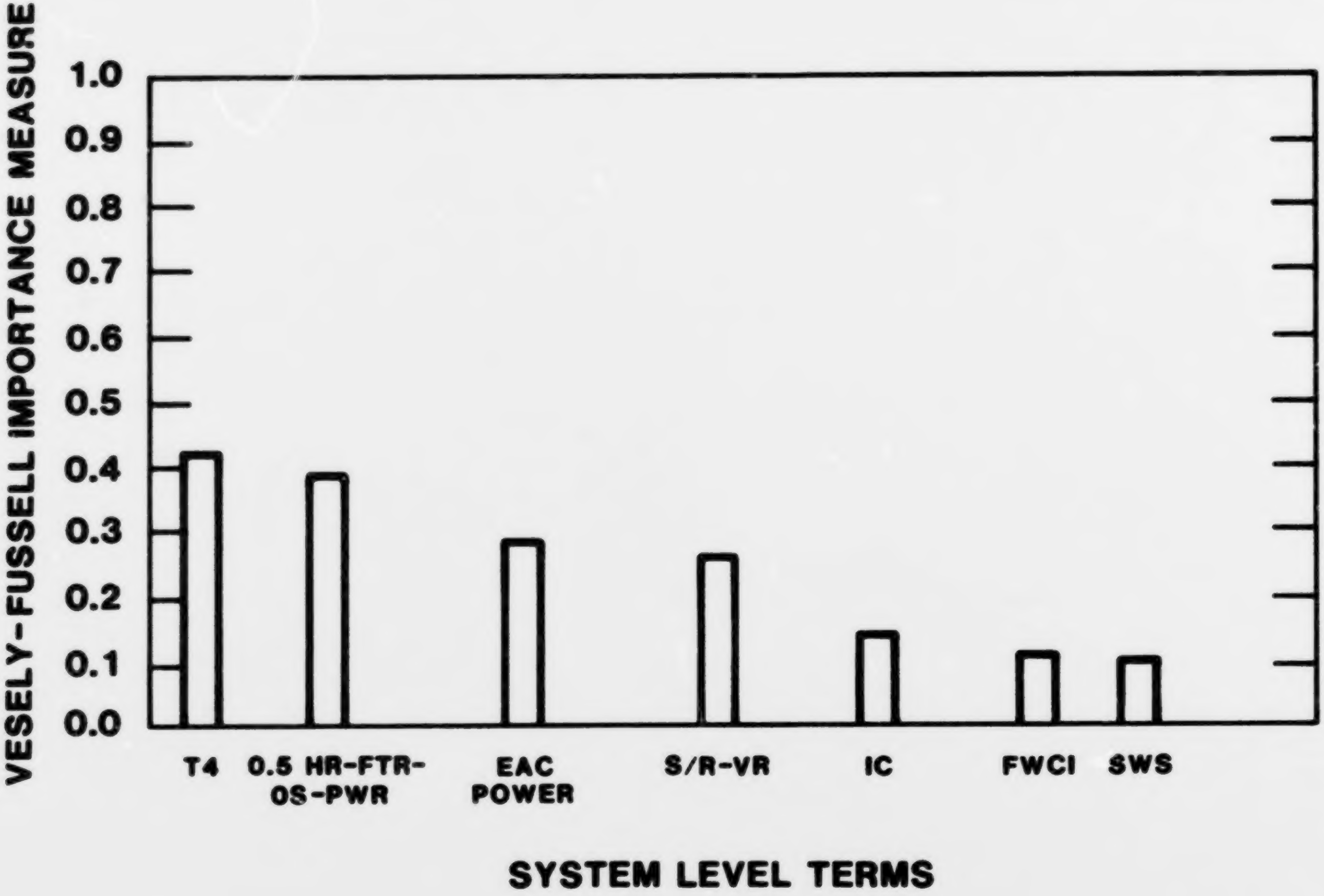

Figure 86. Millstone, Importance of Front Line and Support Systems to Total Core Melt Frequency: Criteria 1 and 2

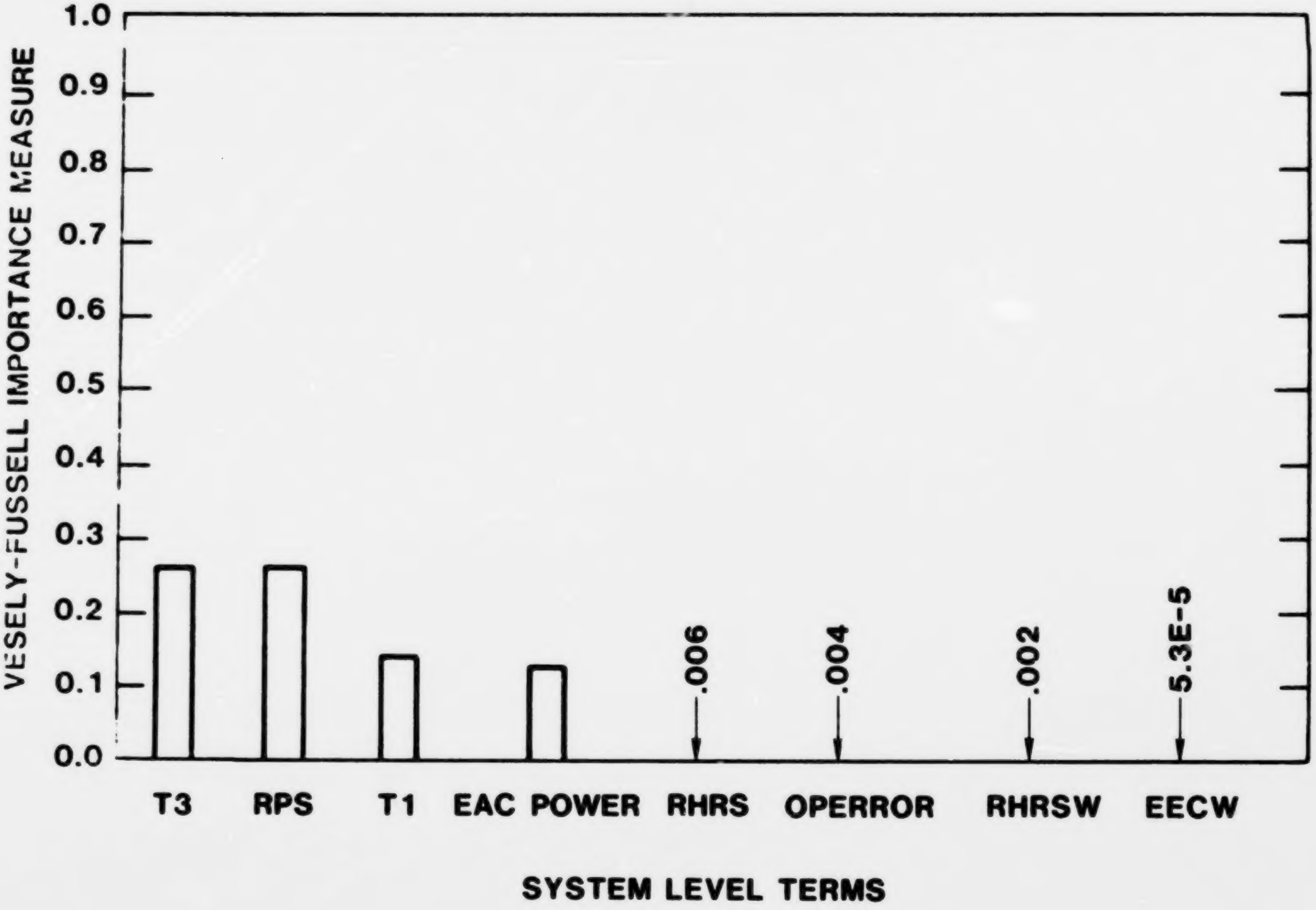

Figure 87. Browns Ferry-1, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

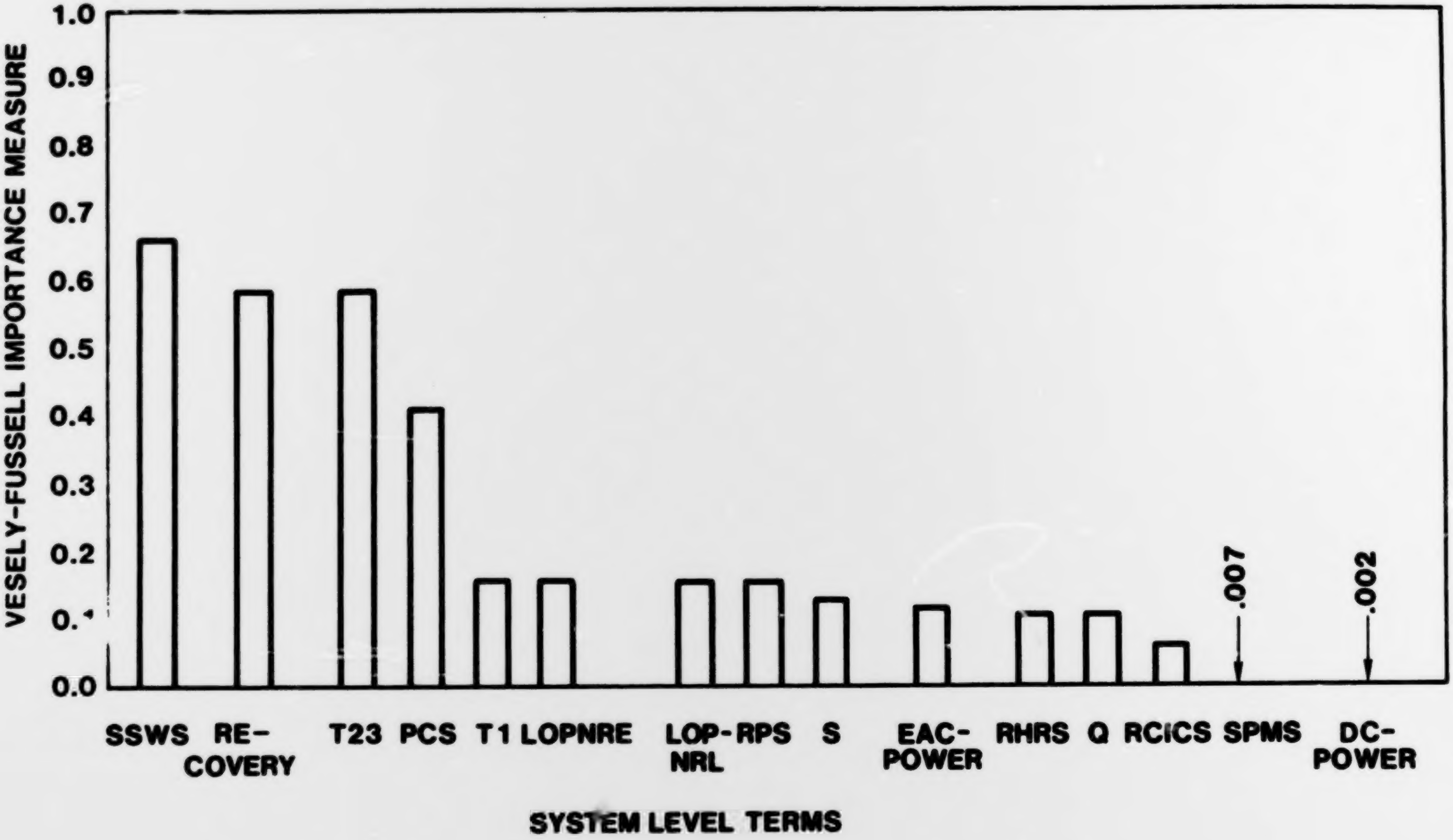

Figure 88. Grand Gulf, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

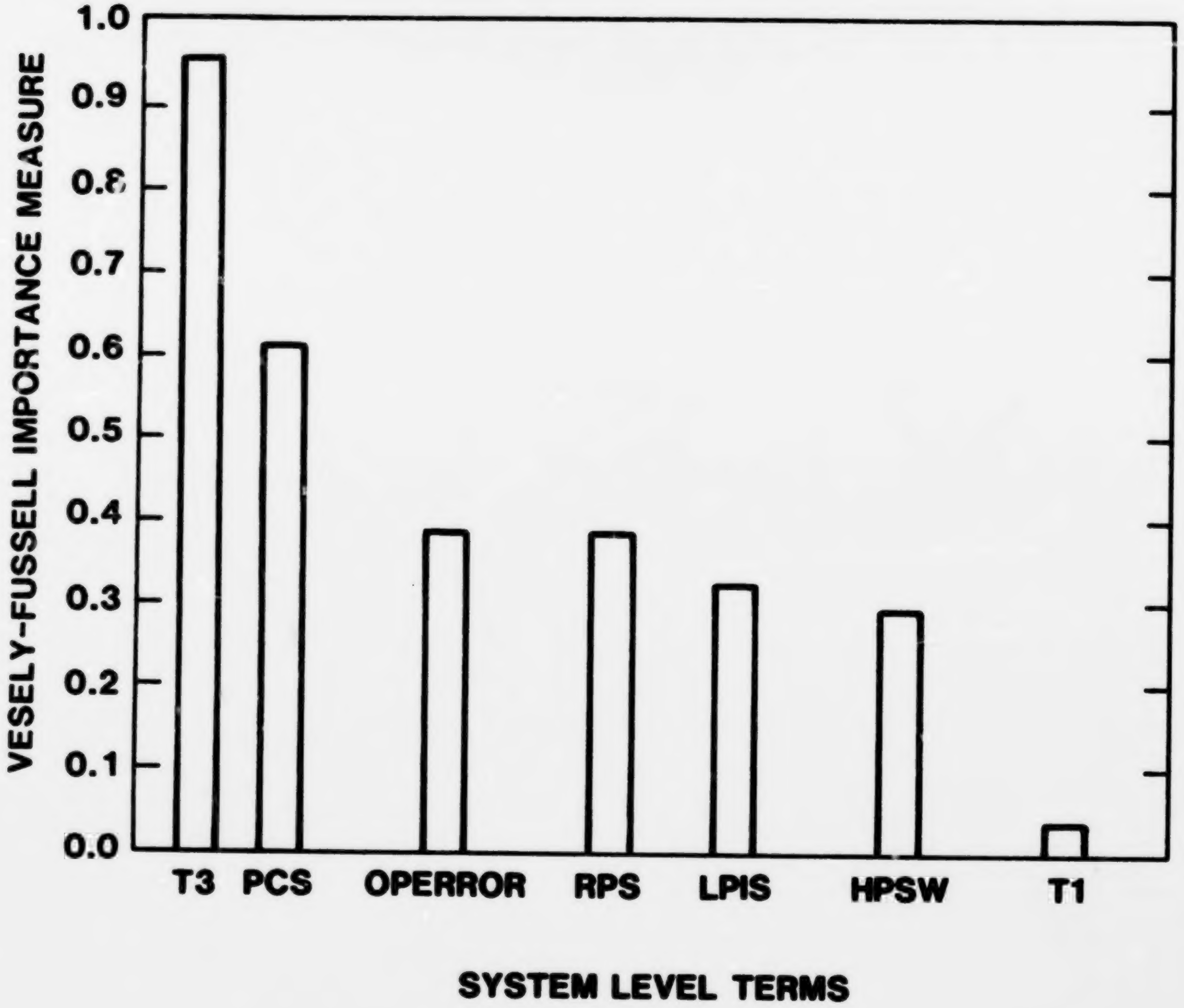

Figure 89. Peach Bottom, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

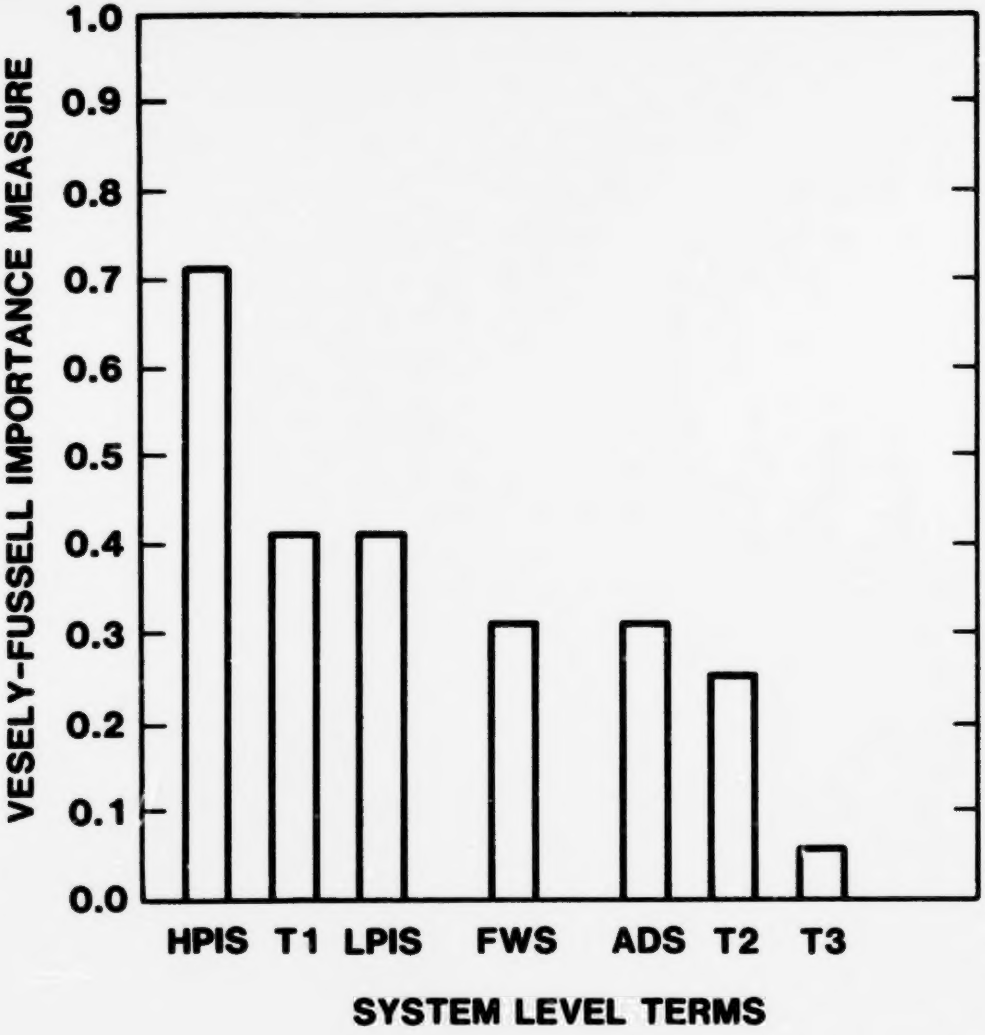

Figure 90. Limerick, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

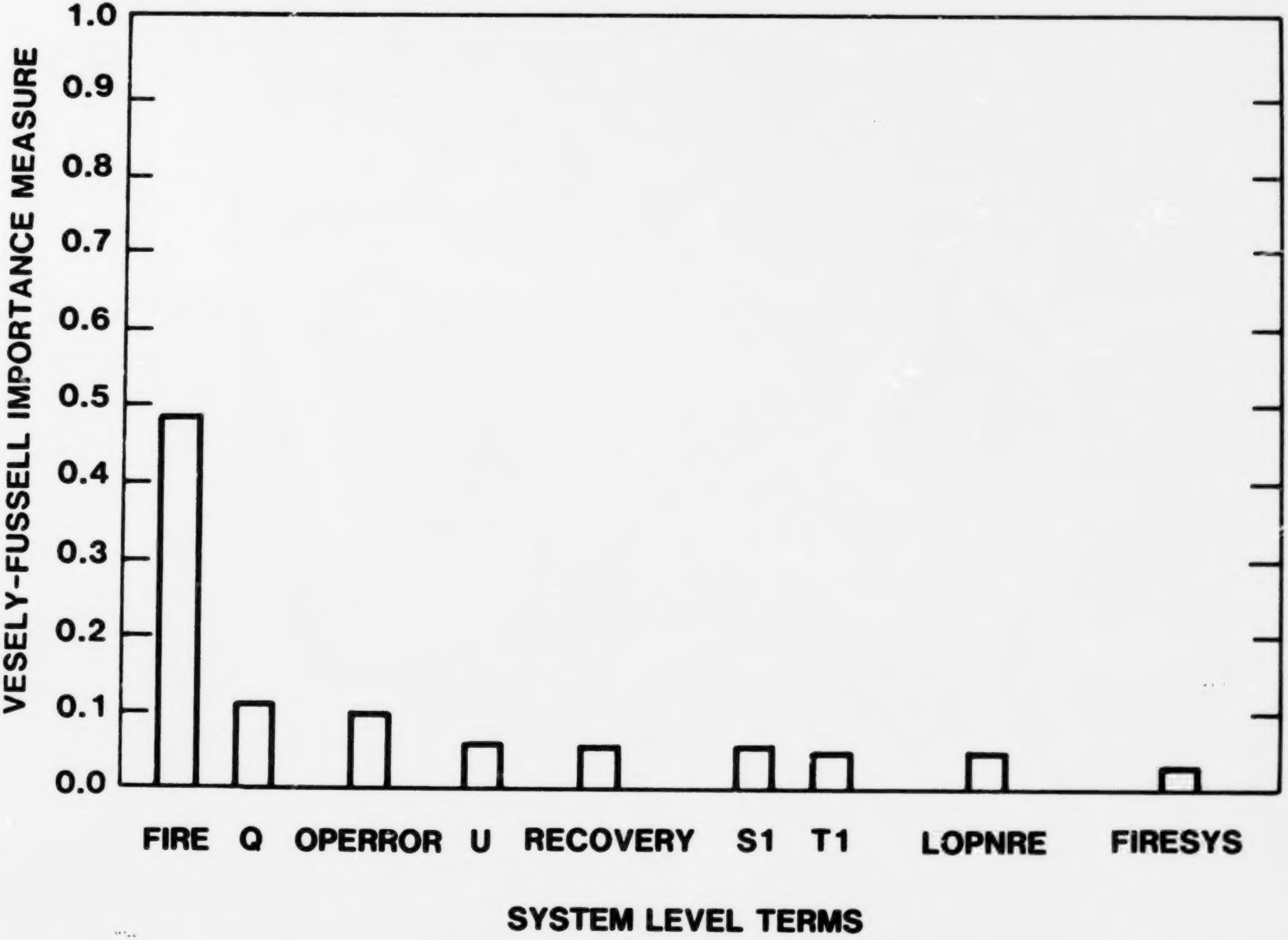

Figure 91. Big Rock Point, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

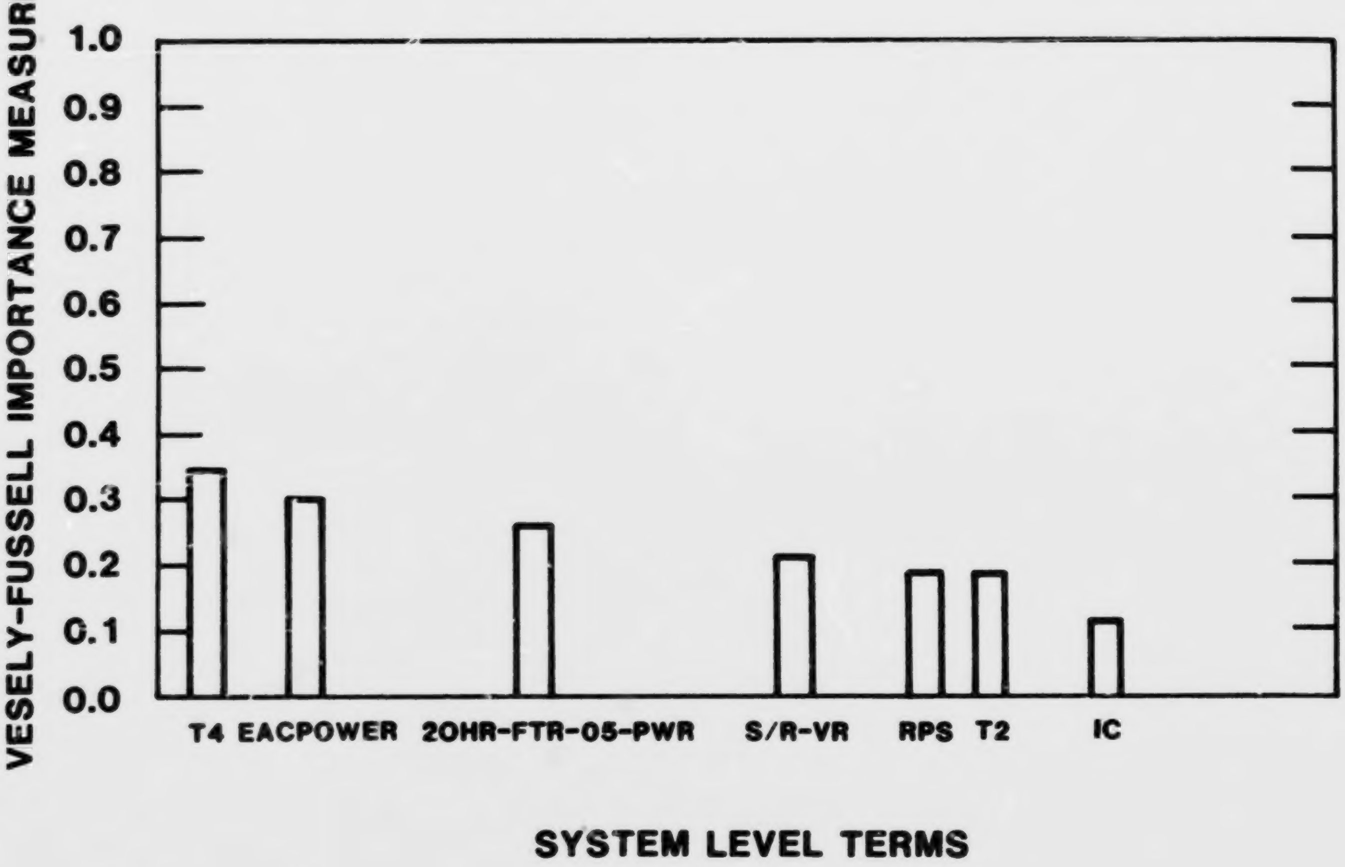

Figure 92. Millstone, Importance of Front Line and Support Systems to Total Plant Risk: Criterion 3

IV. DISCUSSIONS OF RESULTS BETWEEN PLANTS

The results of the various plants indicate that there is no strong trend of similar systems dominating core melt frequency or plant risk across or within manufacturer types. In the four studies which considered external events; Big Rock Point (BRP), Zion, and Indian Point 2 and 3; the external events tend to be significant to core melt frequency, and more significant than plant systems with respect to risk and containment failure. However, the actual external events which are dominant seem to be sensitive to site specific features. The studies which considered external events also used significantly different containment response models than the other PRAs. The Zion and Indian Point models greatly reduce the importance of plant systems to risk compared to the other PRA containment response models.

Operator errors are significant in ten of the fourteen PRAs analyzed. However, the type of operator errors and the level of importance of operator errors vary significantly from plant to plant. Furthermore, it is difficult to achieve an accurate measure of operator importance due to the nature in which unavailabilities developed from the PRAs which did not publish sequence cut sets contain significant human error contributions as well as hardware faults. But the manner in which the PRAs often describe system failures does not permit the human error contribution to be separated from the hardware faults. Even PRAs with published sequence cut sets such as Calvert Cliffs RSSMAP and Crystal River IREP often model human errors into the basic event unavailability.

Another discrepancy between PRAs which impacts the importance calculations is the manner in which recovery actions are modeled. Some PRAs model recovery actions as an explicit basic event. Other PRAs simply multiply a recovery factor with the value of an entire cut set. So the importance of recovery actions cannot always be calculated for PRAs. In general, PRA treatment of recovery models are weak, if recovery is discussed at all. This adds to the difficulty of trying to determine the importance of recovery actions.

The presentation of sequence models in the various PRAs presented significant difficulties for the application of the importance methodology for this analysis. Only six of the PRAs published sequence cut sets derived from sequence Boolean equations. These studies are Grand Gulf RSSMAP, Calvert Cliffs, Millstone RSSMAP, Oconee RSSMAP, Crystal River IREP, and ANO-1 IREP. These studies are very compatible with the importance methodology of this analysis. The results for these plants present a very accurate calculation of the basic event and system importances for that portion of the core melt frequency modeled. However, as much as 30 percent of the cut sets for some of these PRAs were either not published in the PRA or not included in the dominant sequences. The importance results from the other nine PRAs are based on cut sets derived from highly simplified fault trees. For several of the PRAs, the basic events are virtually system and event tree level events. Importance calculations from such cut sets have a qualitative value for demonstrating which systems and events tend to be important for a certain plant. However, the actual quantitative values calculated for such generalized basic events should not be viewed as highly accurate importance measures.

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APPENDIX A

APPENDIX A

DOMINANT SEQUENCES AND CUT SETS

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Abstract to Appendix A

The system level and basic event cut sets of the dominant sequences for each PRA are listed in this appendix. For those PRAs for which simplified fault trees were constructed to develop cut sets, the sequence fault trees are also included. A brief description of the dominant sequences is included as well for each PRA.

A1. Sequoyah RSSMAP Dominant Sequences, System Level and Basic Event Cut sets, and Sequence Fault Trees.

Table A1-1

Sequoyah RSSMAP

Description of Dominant Accident Sequences

Sequence	Description
S_1H	Failure of the Emergency Core Cooling Recirculation System (ECRS) often a small LOCA (2 in. $\leq$ D $\leq$6 in.).
S_2H	Failure of the Emergency Core Cooling Recirculation System often a small LOCA (0.5 in. $\leq$ D $<$ 2 in.).
S_1HF	Failure of the ECRS and the Containment Spray Recirculation System (CSRS) given an S_1-sized small LOCA.
S_2HF	Failure of the ECRS and CSRS given an S_2-sized small LOCA.
S_1D	Failure of the Emergency Core Cooling Injection System (ECIS) given an S_1-sized small LOCA.
S_2D	Failure of the ECIS given an S_2-sized small LOCA.
V	Interfacing systems LOCA, rupture of Low Pressure Injection System (LPIS) check values, causing a LOCA from the primary system through the LPIS.

Table A1-2

Sequoyah RSSMAP

System Level Cut Sets of Significant Sequences

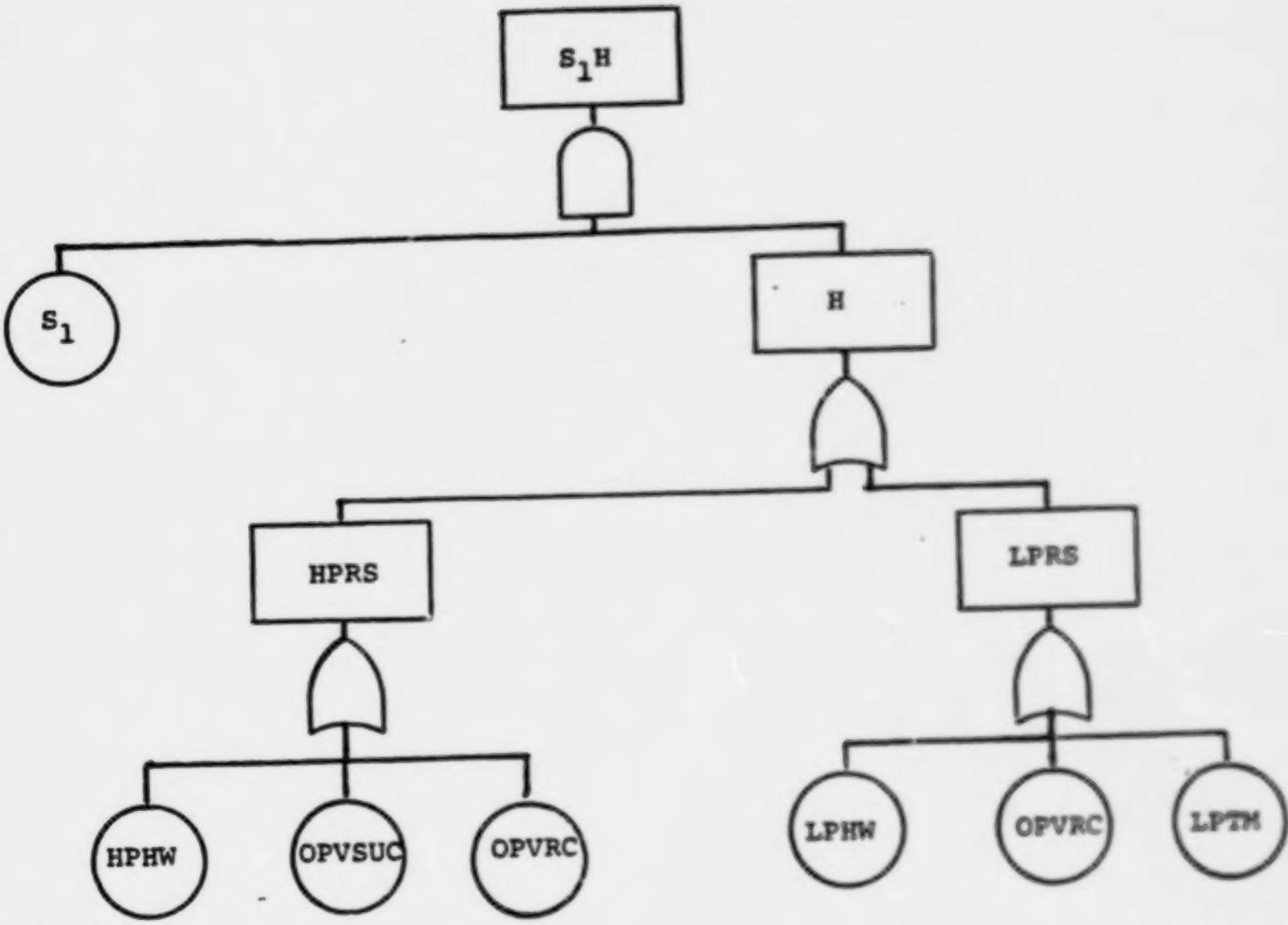
Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
S_1H	S_1*HPRS	2.0E-6
	S_1*OPERROR	2.9E-6
	S_1*LPRS	1.4E-6
	S_1*OPPERROR	2.9E-6
	S_1*LPRS	1.6E-7
S_2H	S_2*HPRS	3.6E-6
	S_2*OPERROR	5.4E-6
	S_2*LPRS	2.5E-6
	S_2*OPERROR	5.4E-6
	S_2*LPRS	2.9E-7
S_1HF	S_1*HPRS*CSRS	6.5E-9
	S_1*OPERROR*CSRS	9.7E-9
	S_1*LPRS*CSRS	4.5E-9
	S_1*OPERROR*CSRS	9.7E-9
	S_1*LPRS*CSRS	5.2E-10
	S_1*OPERROR*CSRS	2.9E-6
S_2HF	S_2*HPRS*CSRS	1.2E-8
	S_2*OPERROR*CSRS	1.8E-8
	S_2*LPRS*CSRS	8.3E-9
	S_2*OPERROR*CSRS	1.8E-8
	S_2*LPRS*CSRS	9.5E-10
	S_2*OPERROR*CSRS	5.4E-6
S_1D	S_1*HPIS	2.7E-6
	S_1*HPIS	6.9E-7
S_2D	S_2*HPIS	5.0E-6
	S_2*HPIS	1.3E-6
V	V	4.6E-6

Table A1-3

Sequoyah RSSMAP

Basic Event Cut Sets of Significant Sequences

Sequence	Cut Sets	Value $(R\text{-}yr)^{-1}$
S_1H	S_1*HPHW	2.0E-6
	S_1*OPVSUC	2.9E-6
	S_1*LPHW	1.4E-6
	S_1*OPVRC	2.9E-6
	S_1*LPTM	1.6E-7
S_2H	S_2*HPHW	3.6E-6
	S_2*OPVSUC	5.4E-6
	S_2*LPHW	2.5E-6
	S_2*OPVRC	5.4E-6
	S_2*LPTM	2.9E-7
S_1HF	S_1*HPHW*CSRS	6.5E-9
	S_1*OPVSUC*CSRS	9.7E-9
	S_1*LPHW*CSRS	4.5E-9
	S_1*OPVRC*CSRS	9.7E-9
	S_1*LPTM*CSRS	5.2E-10
	S_1*CMSD	2.9E-6
S_2HF	S_2*HPHW*CSRS	1.2E-8
	S_2*OPVSUC*CSRS	1.8E-8
	S_2*LPHW*CSRS	8.3E-9
	S_2*OPVRC*CSRS	1.8E-8
	S_2*LPTM*CSRS	9.5E-10
	S_2*CMSD	5.4E-6
S_1D	S_1*CCHW	2.7E-6
	S_1*SIHW	6.9E-7
S_2D	S_2*CCHW	5.0E-6
	S_2*SIHW	1.3E-6
V	V	4.6E-6



S_1H
S_1
H
HPRS
LPRS
HPHW
OPVSUC
OPVRC
LPHW
OPVRC
LPTM

Figure A1-1. Sequence S_1H Fault Tree, Sequoyah RSSMAP

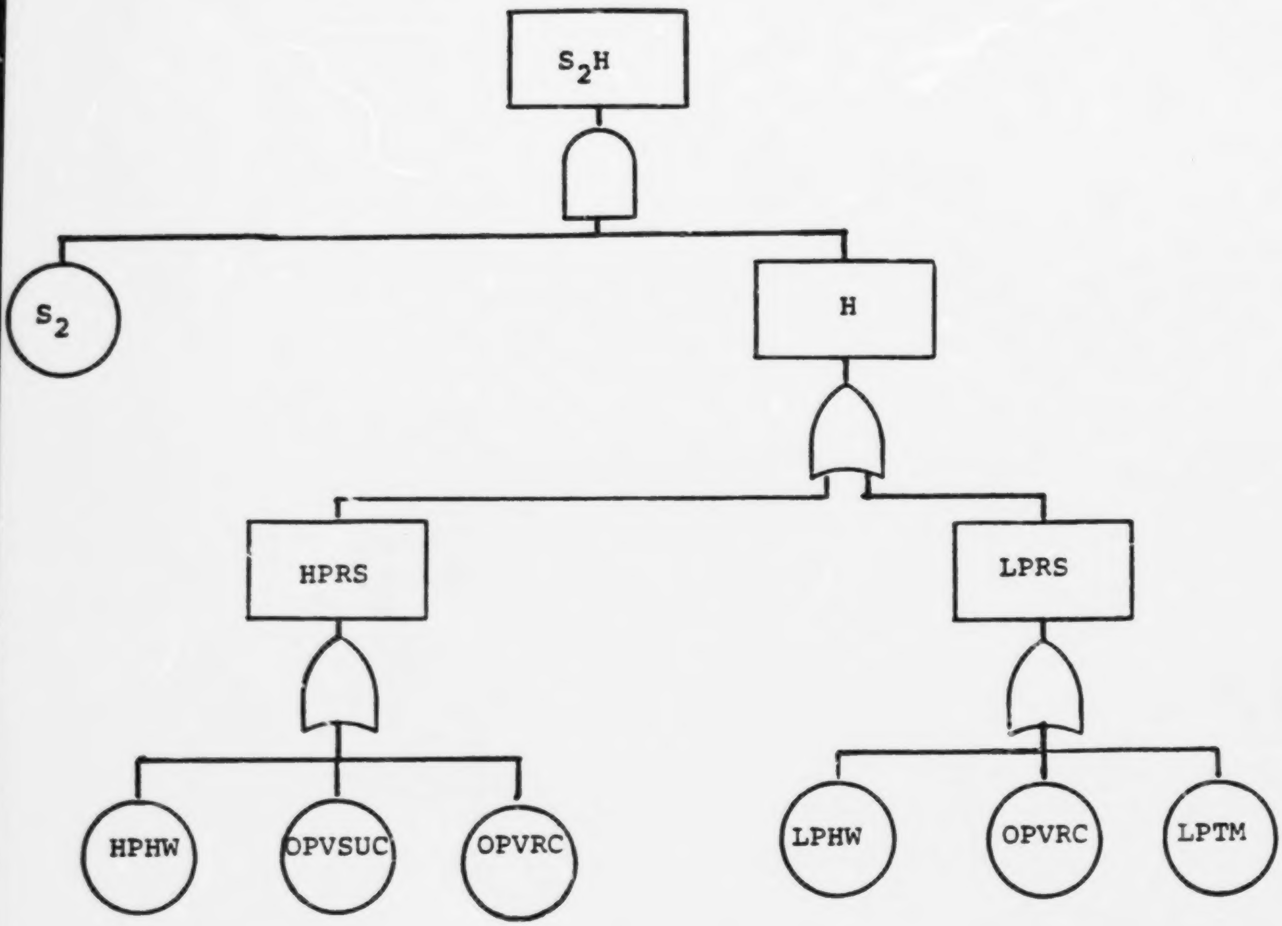

S2H
S2
H
HPRS
LPRS
HPHW
OPVSUC
OPVRC
LPHW
OPVRC
LPTM

Figure A1-2. Sequence S_2H Fault Tree, Sequoyah RSSMAP

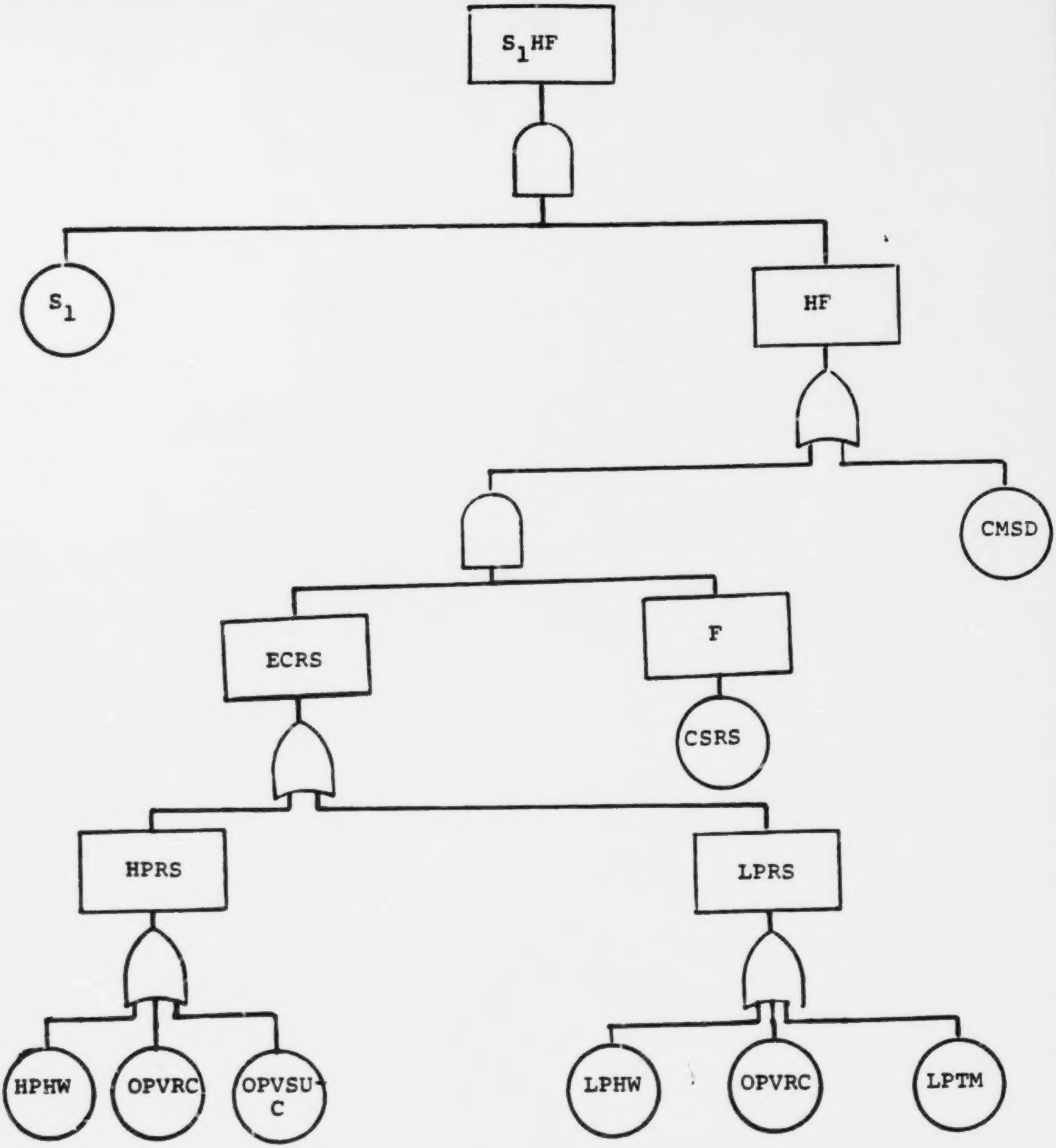

S$_1$HF
S$_1$
HF
CMSD
ECRS
F
CSRS
HPRS
LPRS
HPHW
OPVRC
OPVSU-C
LPHW
OPVRC
LPTM

Figure A1-3. Sequence S$_1$HF Fault Tree, Sequoyah RSSMAP

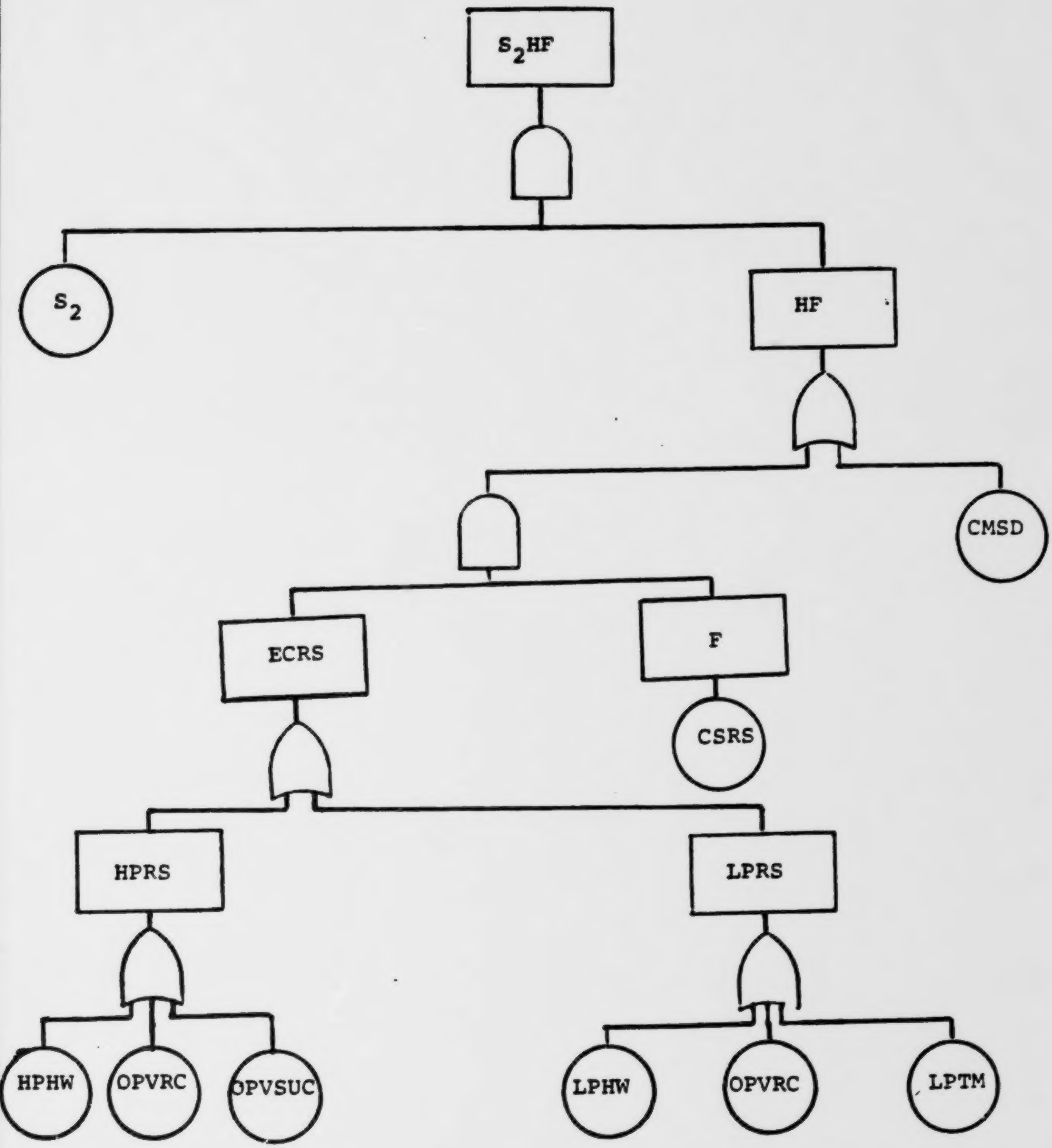

S2HF
S2
HF
CMSD
ECRS
F
CSRS
HPRS
LPRS
HPHW
OPVRC
OPVSUC
LPHW
OPVRC
LPTM

Figure A1-4. Sequence S_2HF Fault Tree, Sequoyah RSSMAP

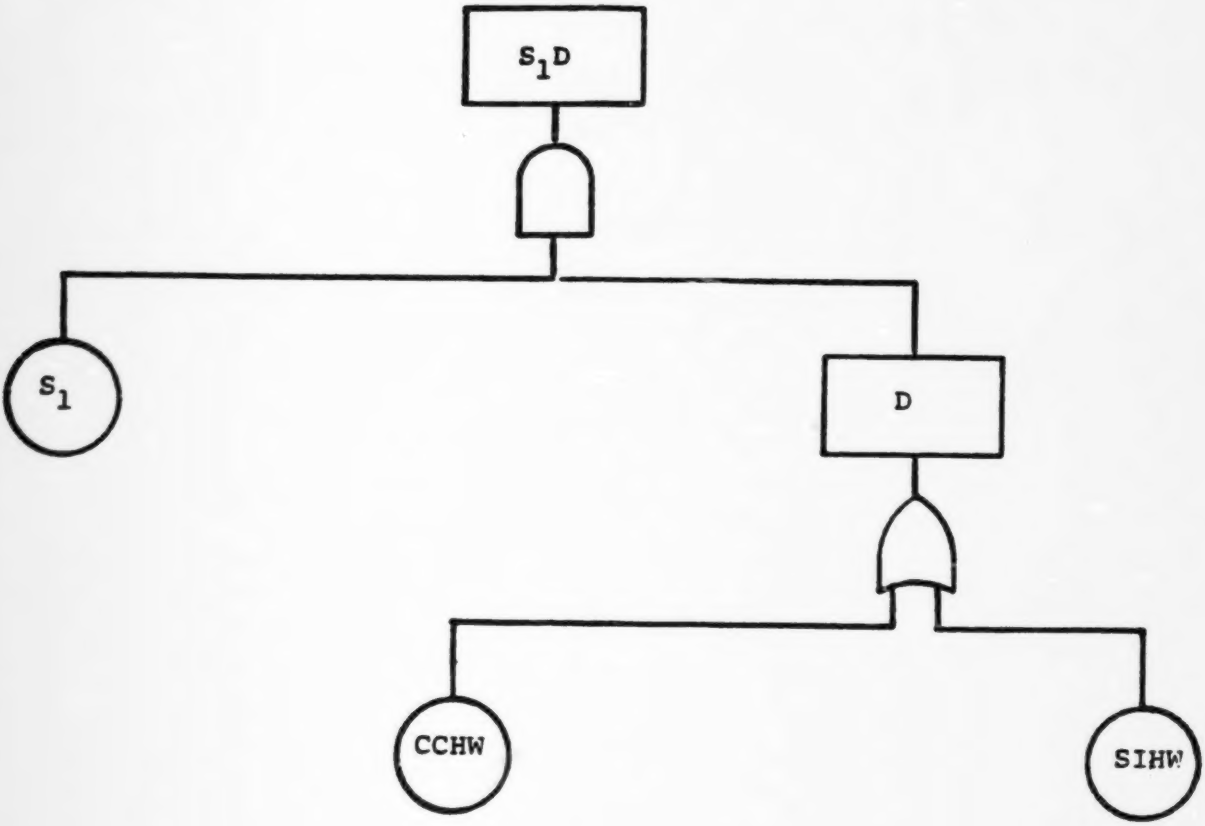

S1D
S1
D
CCHW
SIHW

Figure A1-5. Sequence of S_1D Fault Tree, Sequoyah RSSMAP

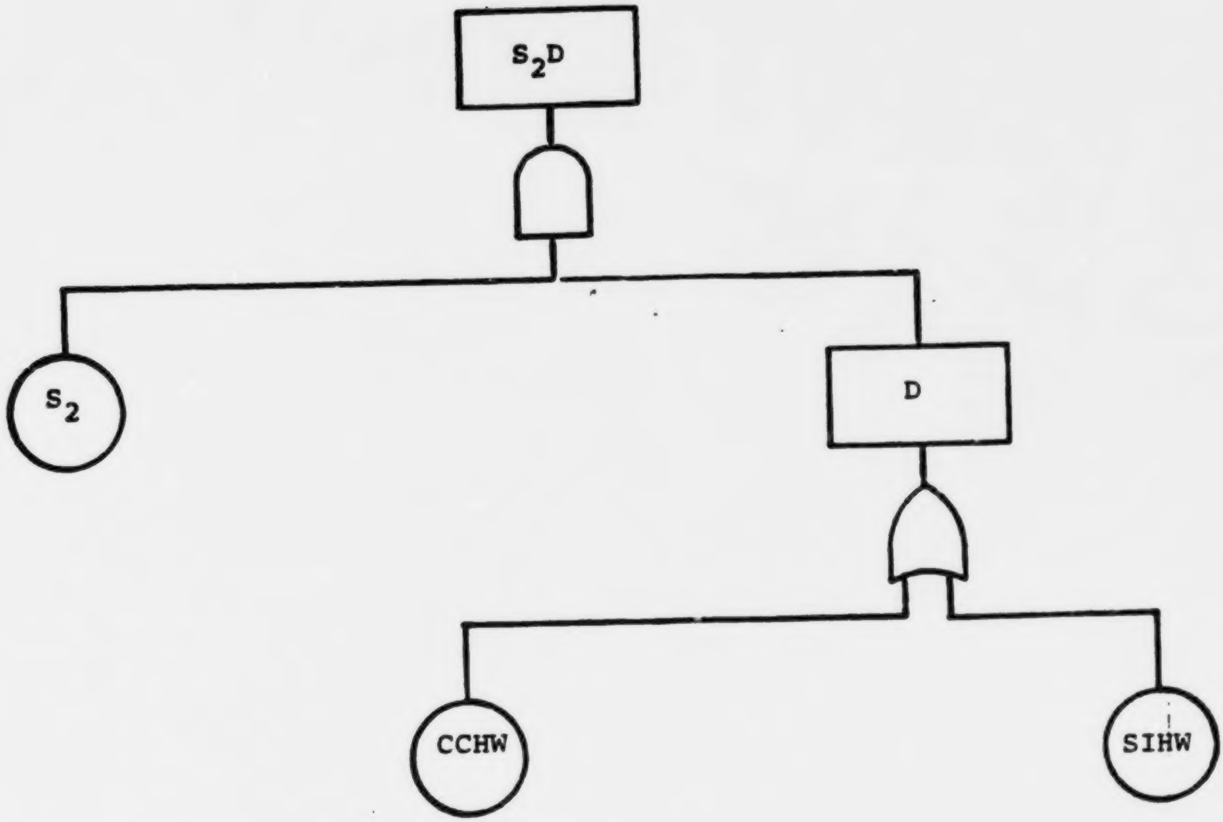

S_2D
S_2
D
CCHW
SIHW

Figure A1-6. Sequence S_2D Fault Tree, Sequoyah RSSMAP

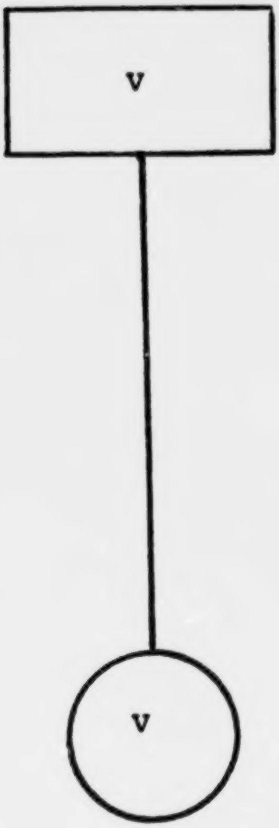


Figure A1-7. Sequence V Fault Tree, Sequoyah RSSMAP

A2. Surry (WASH-1400 PWR) Dominant Sequences, System Level and Basic Event Cut Sets, and Sequence Fault Trees.

Table A2-1

Surry (WASH-1400)

Description of Dominant Accident Sequences

Sequence	Description
TMLB'	Failure of the Power Conversion System (PCS) and the Auxillary Feedwater Systems (AFWS) given a loss of offsite power with a failure to recover onsite or offsite AC power within 3 hours.
V	Interfacing systems LOCA-rupture of the Low Pressure Injection System (LPIS) causing a LOCA from the primary system through the LPIS.
S_2D	Failure of Emergency Core Cooling Injection System (ECIS) given an S_2-sized small LOCA ($0.5 < D < 2$ in.).
S_1H	Failure of the Emergency Core Cooling Recirculation System (ECRS) given an S_1-sized LOCA ($2 < D < 6$ in.).
S_1D	Failure of the ECIS given an S_1-sized small LOCA.
S_2H	Failure of the ECRS given an S_2-sized small LOCA.
TML	Failure of the PCS and AFWS given a transient event.
TKQ	Failure to trip and failure of the Primary System Safety Relief Values to reclose given a transient event.
S_2C	Failure of the Containment Spray Injection System given an S_2-sized small LOCA.

Table A2-2

Surry (WASH-1400)
System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
TMLB'	T_1*LOPNRE*EACPOWER*AFWS *RECOVERY*LOPNRL	3.0E-6
V	V	4.0E-6
S_2D	S_2*HPIS	8.6E-6
S_1H	S_1*LPRS	1.5E-6
	S_1*HPRS	1.5E-6
S_1D	S_1*ACC	2.5E-7
	S_1*HPIS	2.6E-6
S_2H	S_2*LPRS	1.5E-6
	S_2*HPRS	1.5E-6
TML	T_1*LOPNRE*EACPOWER*AFWS	6.0E-6
	T_2*RECOVERY*AFWS	9.8E-7
TKQ	T_3*RPS*RV	3.0E-6
S_2C	S_2*CSIS	2.0E-6

Table A2-3

Surry (WASH-1400)
Basic Event Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
TMLB'	TLOP*NROP1*EMAC*AFWST *NREMAC*NROP3	3.0E-6
V	V	4.0E-6
S_2D	S_2*HPIS	8.6E-6
S_1H	S_1*HLPRS S_1*HPRS	1.5E-6 1.5E-6
S_1D	S_1*ACC S_1*HPIS	2.5E-7 2.6E-6
S_2H	S_2*HLPRS S_2*HPRS	1.5E-6 1.5E-6
TML	TFW*RLOP*NROP1*EMAC*AFWST	6.0E-6
	TFW*$\overline{\text{RLOP}}$*NRMFW1*EMAC*AFWS	9.8E-7
TKQ	T*RPS*RV	3.0E-6
S_2C	S_2*CSIS	2.0E-6

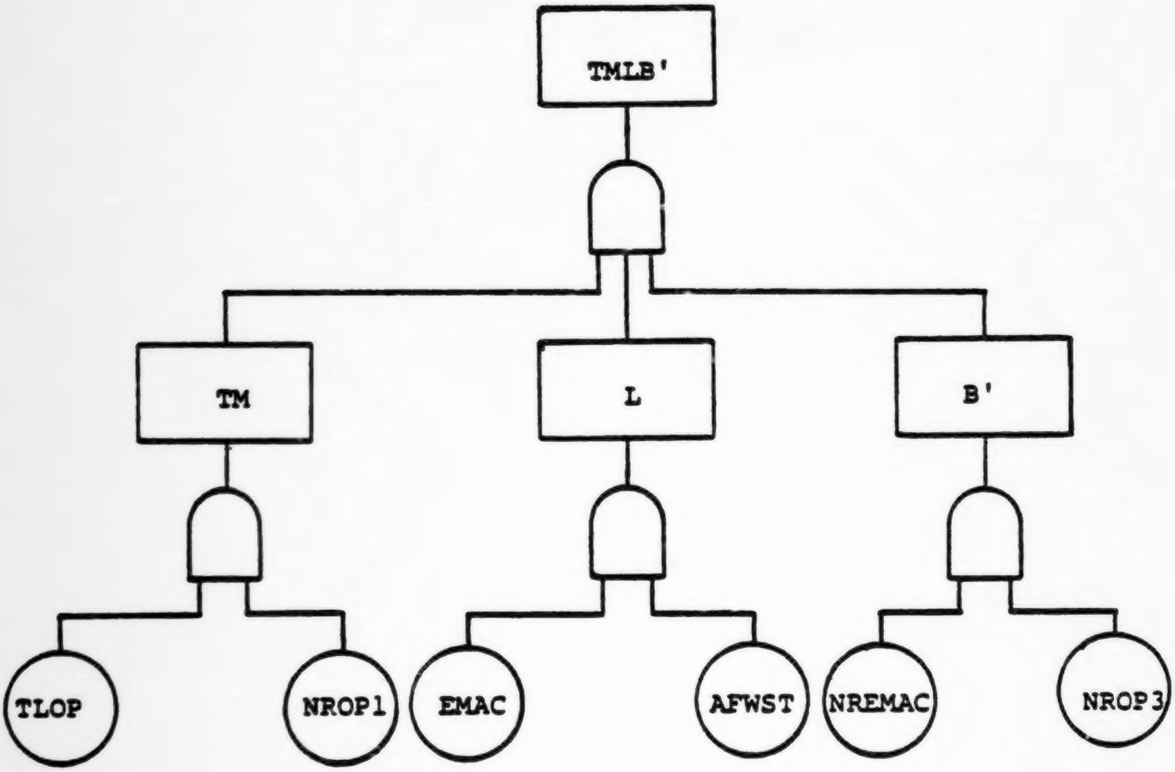

TMLB'
TM
L
B'
TLOP
NROP1
EMAC
AFWST
NREMAC
NROP3

Figure A2-1. RSS PWR Fault Tree for TMLB' Event Sequence

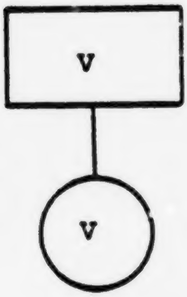


Figure A2-2. RSS PWR Fault Tree for V Event Sequence

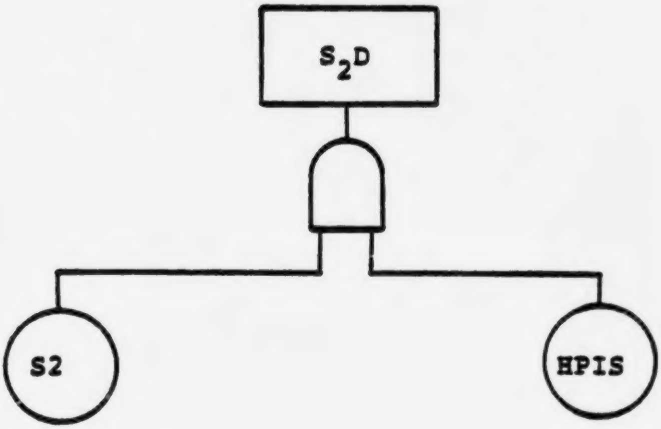


Figure A2-3. RSS PWR Fault Tree for S_2D Event Sequence

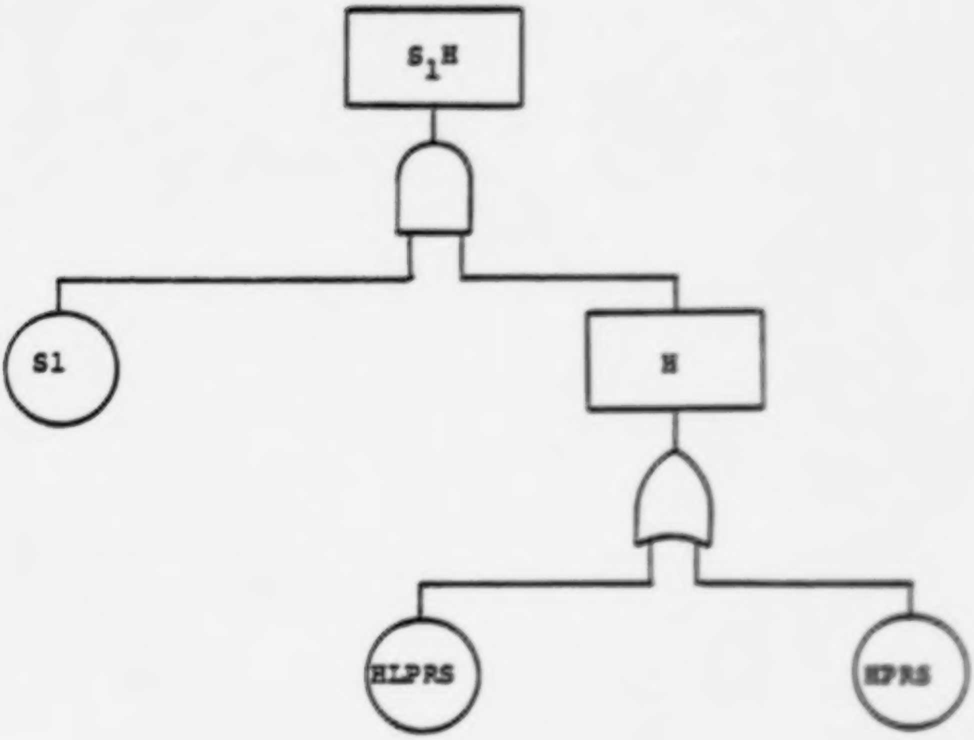

S_1H
S1
H
HLPRS
HPRS

Figure A2-4. RSS PWR Fault Tree for S_1H Event Sequence

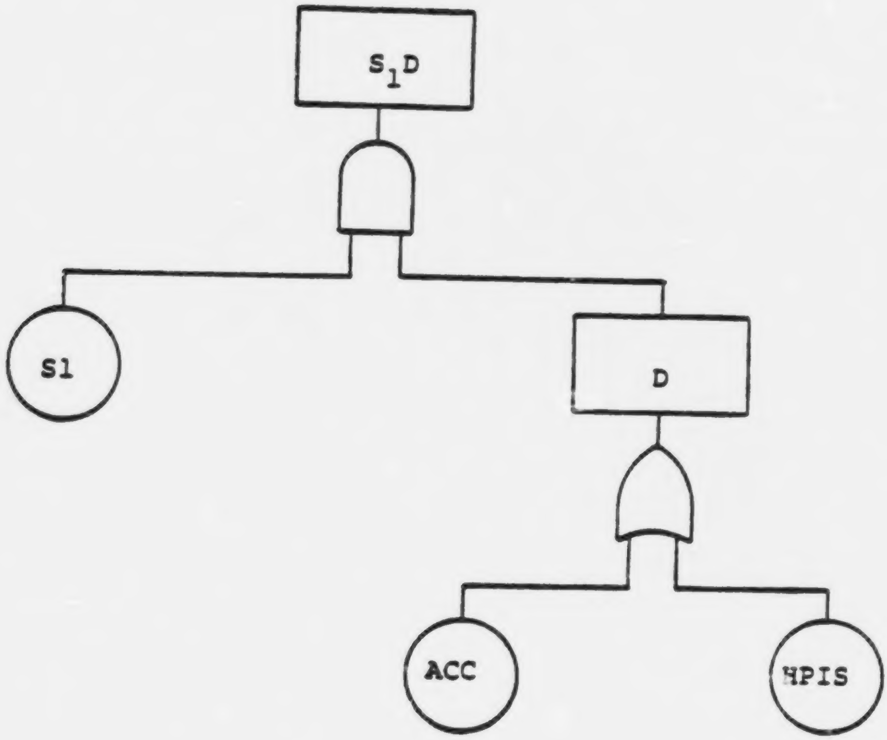

S_1D
S1
D
ACC
HPIS

Figure A2-5. RSS PWR Fault Tree for S_1D Event Sequence

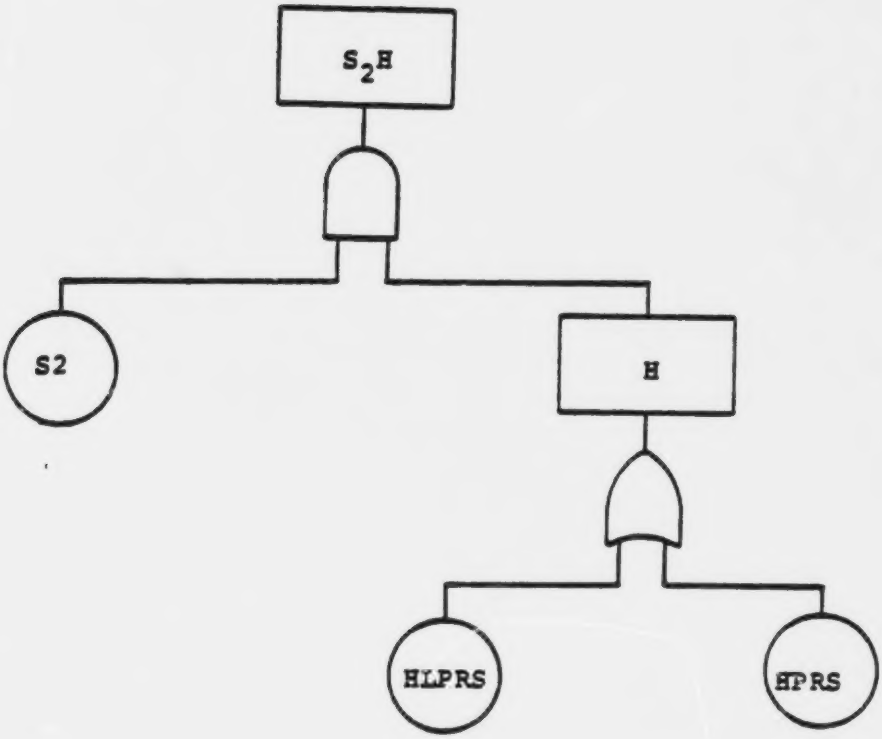

S_2H
S2
H
HLPRS
HPRS

Figure A2-6. RSS PWR Fault Tree for S_2H Event Sequence

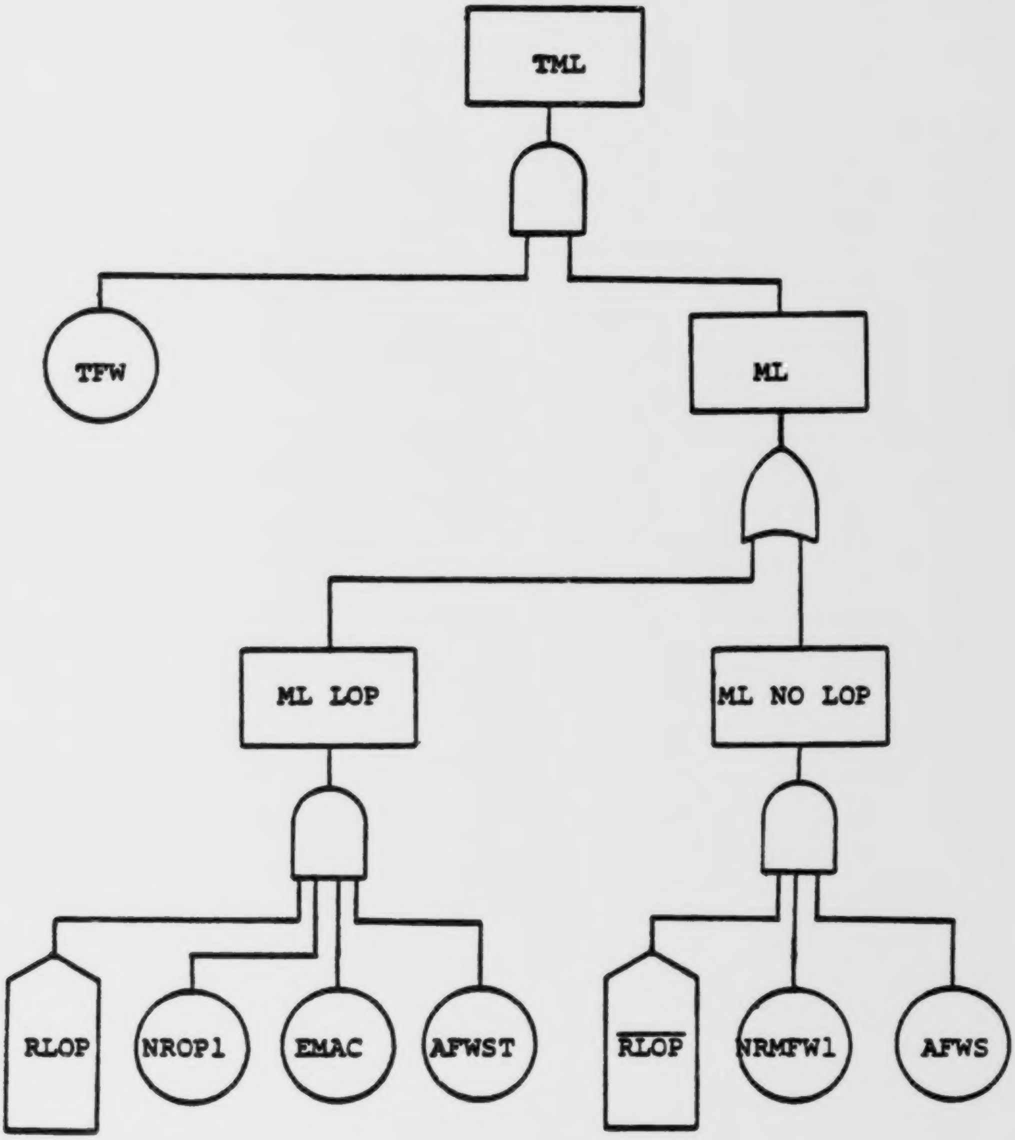

TML
TFW
ML
ML LOP
ML NO LOP
RLOP
NROP1
EMAC
AFWST
$\overline{\text{RLOP}}$
NRMFW1
AFWS

Figure A2-7. RSS PWR Fault Tree for TML Event Sequence

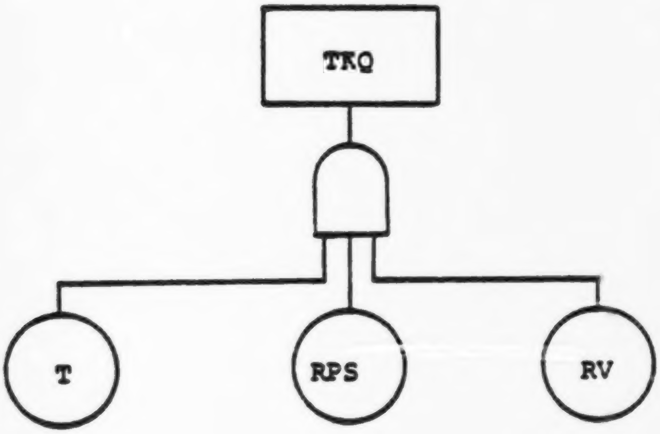


Figure A2-8. RSS PWR Fault Tree for TKQ Event Sequence

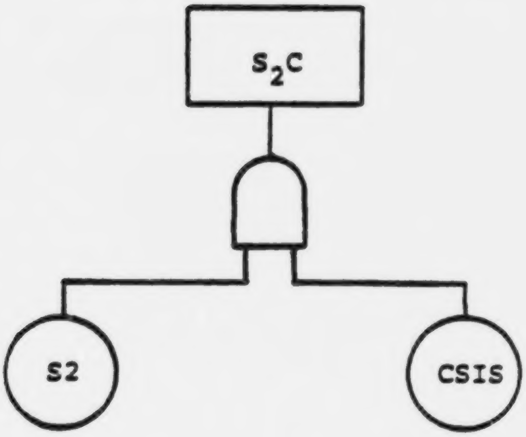


Figure A2-9. RSS PWR Fault Tree for S_2C Event Sequence

A3. Zion Dominant Sequences, System Level and Basic Event Cut sets, and Sequence Fault Trees.

Table A3-1

Zion 1 and 2

Description of Dominant Sequences

Sequence[*]	Description
SLRCIRC	Failure of Recirculation Cooling given a small LOCA.
SESEIS	Loss of all AC Power due to a seismic event.
ALRCIRC	Failure of Recirculation Cooling given a large LOCA.
MLRCIRC	Failure of Recirculation Cooling given a medium LOCA.

---

[*] The Zion sequences correspond to sequences 1,2,3, and 4 of Table 8.10-3 of the Zion Probabilistic Safety Study.

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Table A3-2

Zion 1 and 2

System Level Cut Sets of Significant Sequences

Sequence*	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
1. SLRCIRC	·S_2*OPERROR	4.6E-6
	·S_2*OPERROR	1.2E-6
	·S_2*RHRS	1.0E-6
	·S_2*SUMP	1.0E-6
	·S_2*RHRS*HPRS*OPERROR	3.5E-10
	·S_2*HPRS	2.2E-6
	·S_2*RHRS	2.2E-6
	·S_2*RHRS	2.2E-6
2. SESEIS	SESEIS	5.6E-6
3. ALRCIRC	A*OPERROR	3.8E-6
	A*OPERROR	8.0E-7
	A*OPERROR*RHRS	1.0E-7
	A*RHRS	9.2E-8
	A*RHRS	7.1E-8
4. MLRCIRC	S_1*OPERROR	3.8E-6
	S_1*OPERROR	8.0E-7
	S_1*OPERROR*RHRS	1.0E-7
	S_1*RHRS	9.2E-8
	S_1*RHRS	7.1E-8

---

*The Zion sequences correspond to sequences 1, 2, 3, and 4 of Table 8.10-3 of the Zion Probabilistic Safety Study.

Table A3-3

Zion 1 and 2

Basic Event Cut Sets of Significant Sequences

Sequence*	Cut Sets	Frequency (R-yr)$^{-1}$
1. SLRCIRC	S*SOPSWITCH	4.6E-6
	S*SOPRWST	1.2E-6
	S*CMRHRAB	1.0E-6
	S*SUMP	1.0E-6
	S*SMULTHW	3.5E-10
	S*MANSICH	2.2E-6
	S*SMANRHRB	2.2E-6
	S*SMANRHRA	2.9E-6
2. SESEIS	SESEIS	5.6E-6
3. ALRCIRC	A*OPSWITCH	3.7E-6
	A*CC9412AB	8.0E-7
	A*MULTHW	1.0E-7
	A*MANTRHR	9.2E-8
	A*RHRAB	7.1E-8
4. MLRCIRC	M*OPSWITCH	3.8E-6
	M*CC9412AB	8.0E-7
	M*MULTHW	1.0E-7
	M*MANTRHR	9.2E-8
	M*RHRAB	7.1E-8

---

*The Zion sequences correspond to sequences 1, 2, 3, and 4 of Table 8.10-3 of the Zion Probabilistic Safety Study.

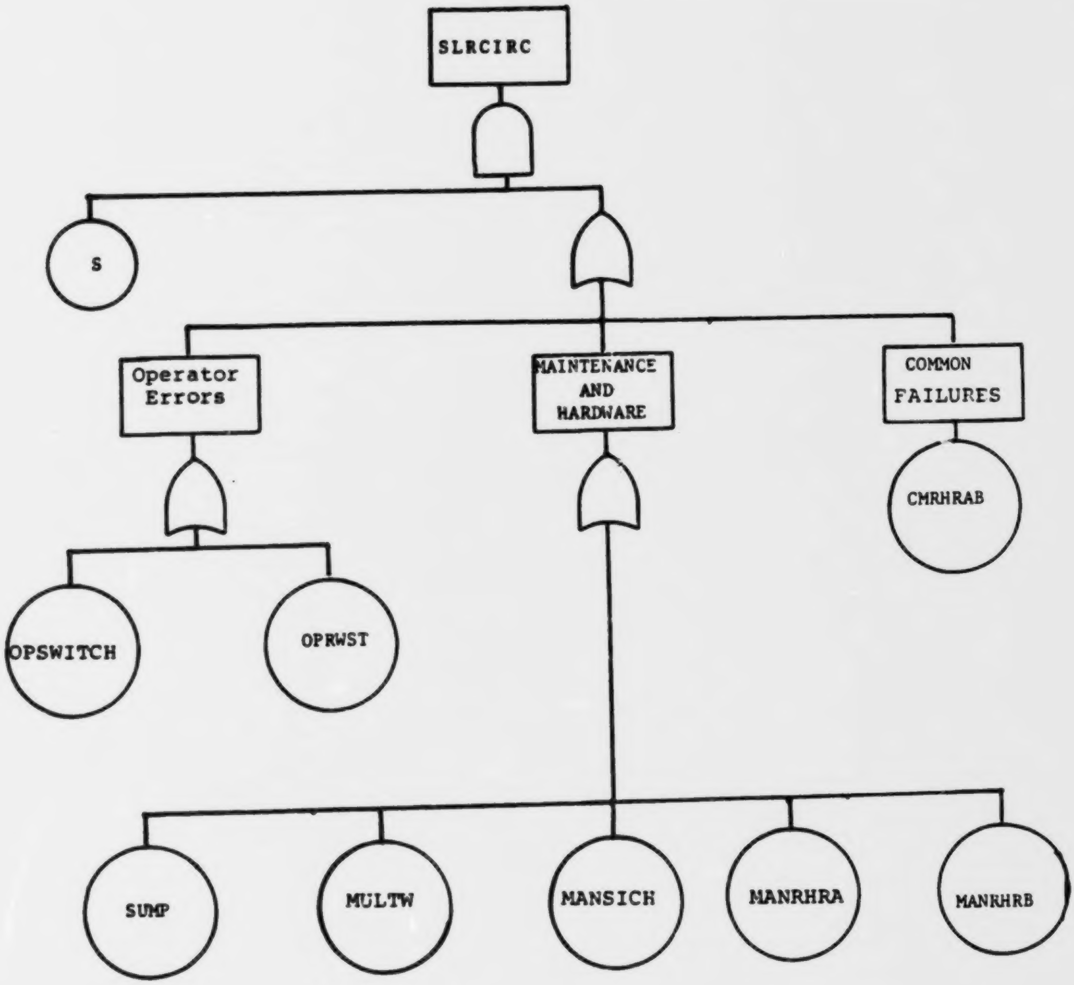
SLRCIRC
S
Operator Errors
MAINTENANCE AND HARDWARE
COMMON FAILURES
CMRHRAB
OPSWITCH
OPRWST
SUMP
MULTW
MANSICH
MANRHRA
MANRHRB

Figure A3-1. Sequence SLRCIRC Fault Tree, Zion.

SESEIS
SESEIS

Figure A3-2. Sequence SESEIS Fault Tree, Zion.

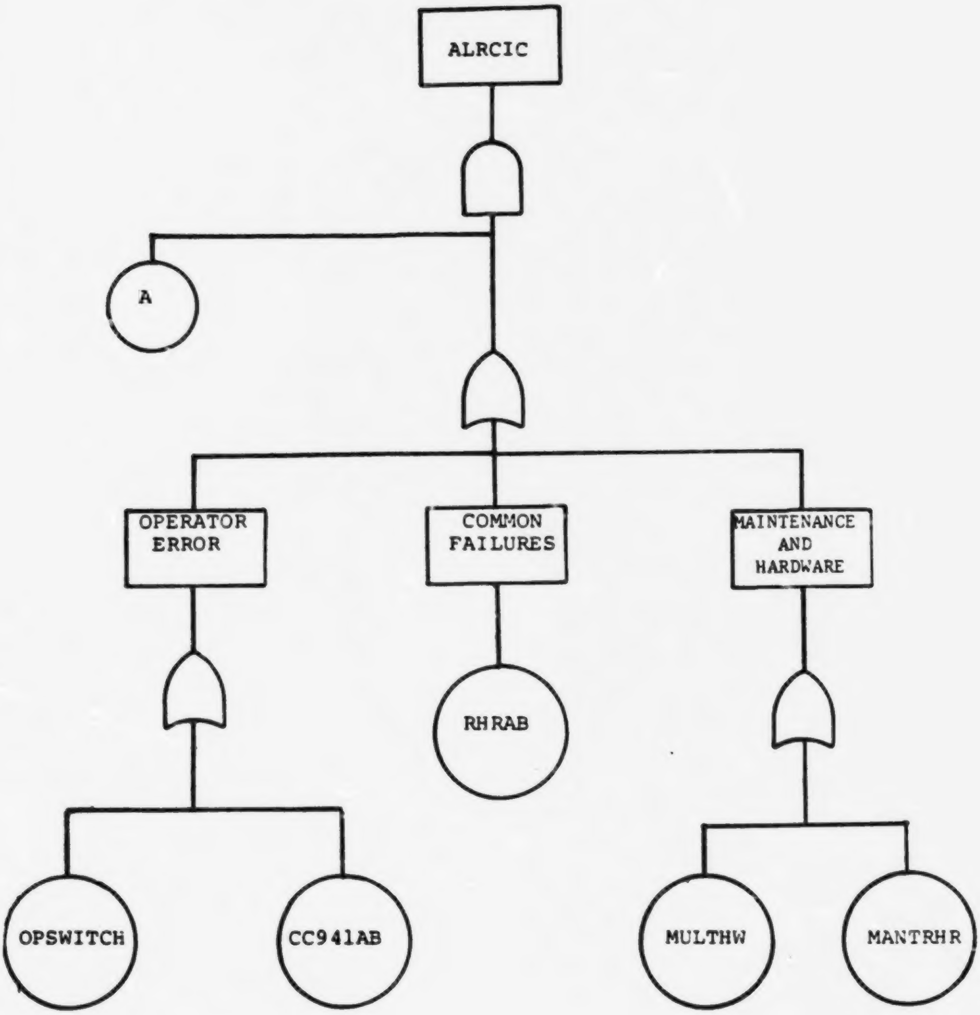
ALRCIC
A
OPERATOR ERROR
COMMON FAILURES
MAINTENANCE AND HARDWARE
RHRAB
OPSWITCH
CC941AB
MULTHW
MANTRHR

Figure A3-3. Sequence ALRCIC Fault Tree, Zion.

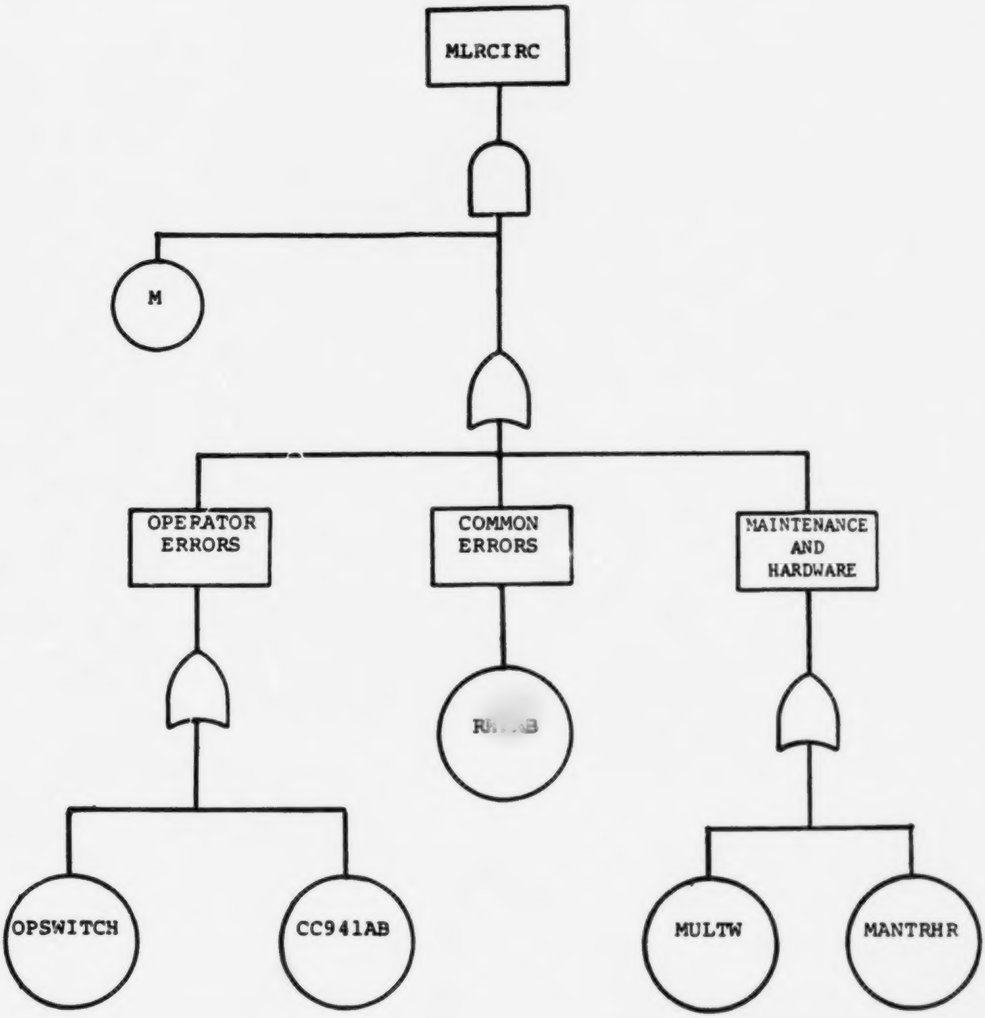
MLRCIRC
M
OPERATOR ERRORS
COMMON ERRORS
MAINTENANCE AND HARDWARE
R[illegible]B
OPSWITCH
CC941AB
MULTW
MANTRHR

Figure A3-4. Sequence MLRCIRC Fault Tree, Zion.

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A4. Indian Point 2 Dominant Sequences, System Level and Basic Event Cut sets, and Sequence Fault Trees.

Table A4-1

Indian Point 2

Description of Dominant Accident Sequences

Sequence[*]	Description
SESEIS	Loss of control or power due to seismic event.
SEFIRE	Specific fires in electrical tunnel and switchgear room causes RCP Seal LOCA and failure of power cables to the Safety Injection Pumps, containment Spray Pumps, and Fan Coolers.
SLFFIR	Specific fires in electrical tunnel cause RCP Seal LOCA and failure of power cables to all MCCS, Safety Injection Pumps, RHR pumps, and containment Spray Pumps.
SEFCTT	Loss of offsite power with failure of two Diesel Generators, RCP Seal LOCA, and failure to recover external AC power within one hour.
SEHUR	Loss of all AC power due to high winds.

---

[*] The Indian Point 2 sequences correspond to sequences 1, 2, 3, 4, and 5 of Table 8-3-9 of the Indian Point PSS.

Table A4-2

Indian Point 2
System Level Cut Sets of Significant Sequences

Sequence*	Cut Sets	Frequency (R-yr)$^{-1}$
SESEIS	SESEIS	1.4E-4
SEFIRE	FIRE	5.6E-5
	FIRE	5.6E-5
	FIRE	3.2E-5
SLFFIR	FIRE	2.4E-5
	FIRE	2.4E-5
SEFCTT	T_1*EACPOWER	1.6E-5
	T_1*EACPOWER	1.6E-5
SEHUR	HURRICANE	2.7E-5

---

*The Indian Point 2 sequences correspond to sequences 1, 2, 3, 4, and 5 of Table 8.3-9 of the Indian Point PSS.

Table A4-3

Indian Point 2
Basic Event Cut Sets of Significant Sequences

Sequence*	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
SESEIS	LOP	1.4E-4
SEFIRE	ETPAB	5.6E-5
	SWGEAR	5.6E-5
	ETCB	3.2E-5
SLFFIR	ETRSPAB	2.4E-5
	ETRSCB	2.4E-5
SEFCTT	TT*FREP5A	1.6E-5
	TT*FREP6A	1.6E-5
SEHUR	HUR	2.7E-5

---

*The Indian Point 2 sequences correspond to sequences 1, 2, 3, 4, and 5 of Table 8.3-9 of the Indian Point PSS.

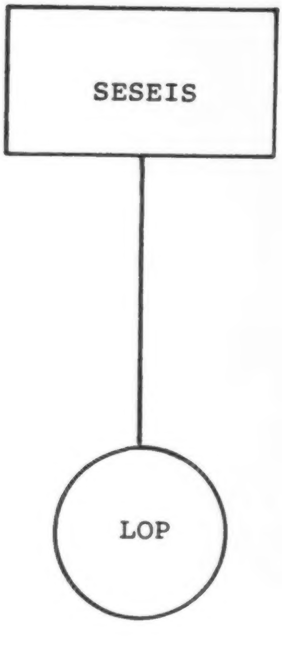

Figure A4-1. Sequence SESEIS Fault Tree, Indian Point 2

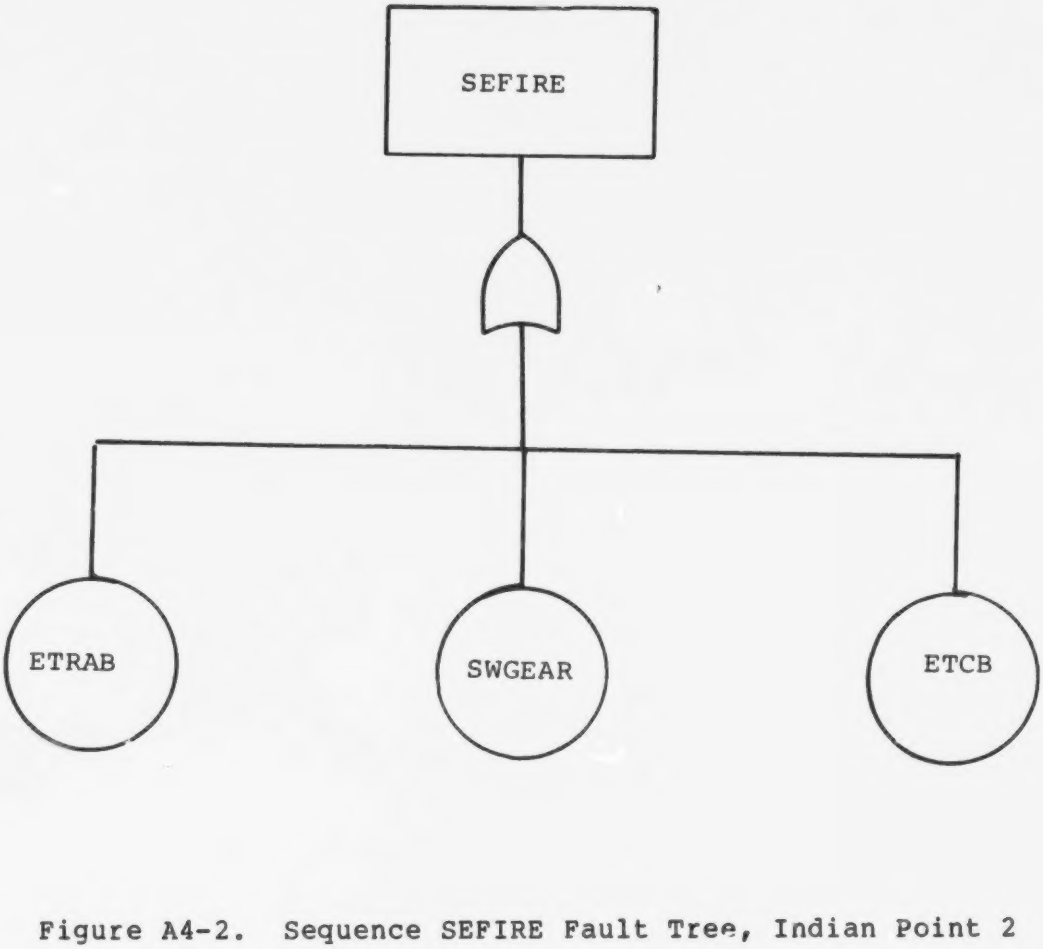
SEFIRE
ETRAB
SWGEAR
ETCB

Figure A4-2. Sequence SEFIRE Fault Tree, Indian Point 2

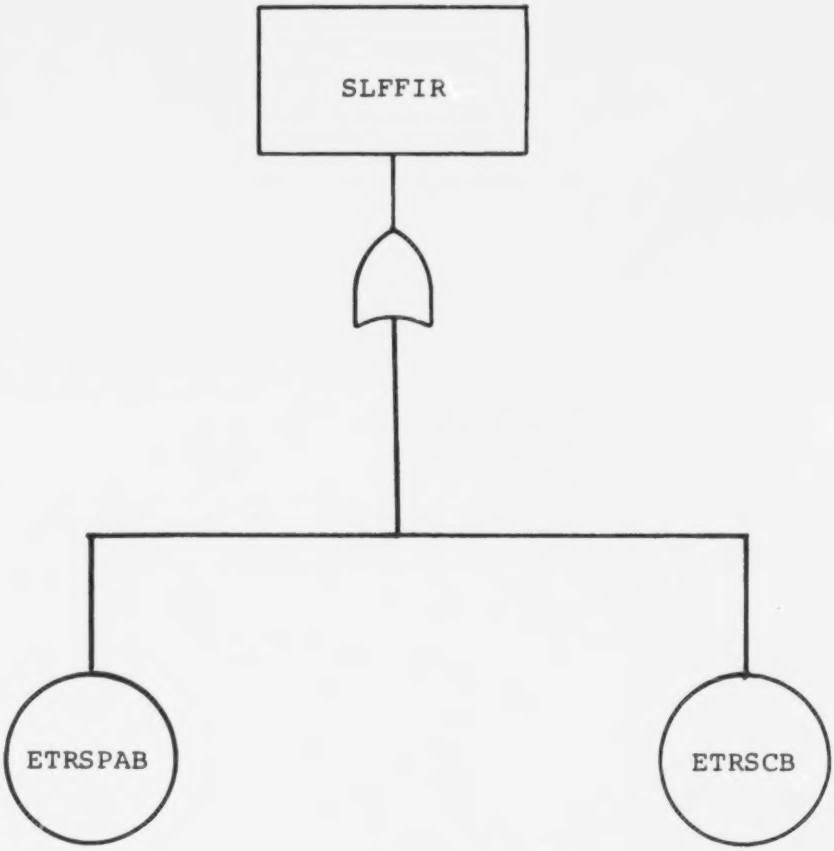
SLFFIR
ETRSPAB
ETRSCB

Figure A4-3. Sequence SLFFIR Fault Trees, Indian Point 2

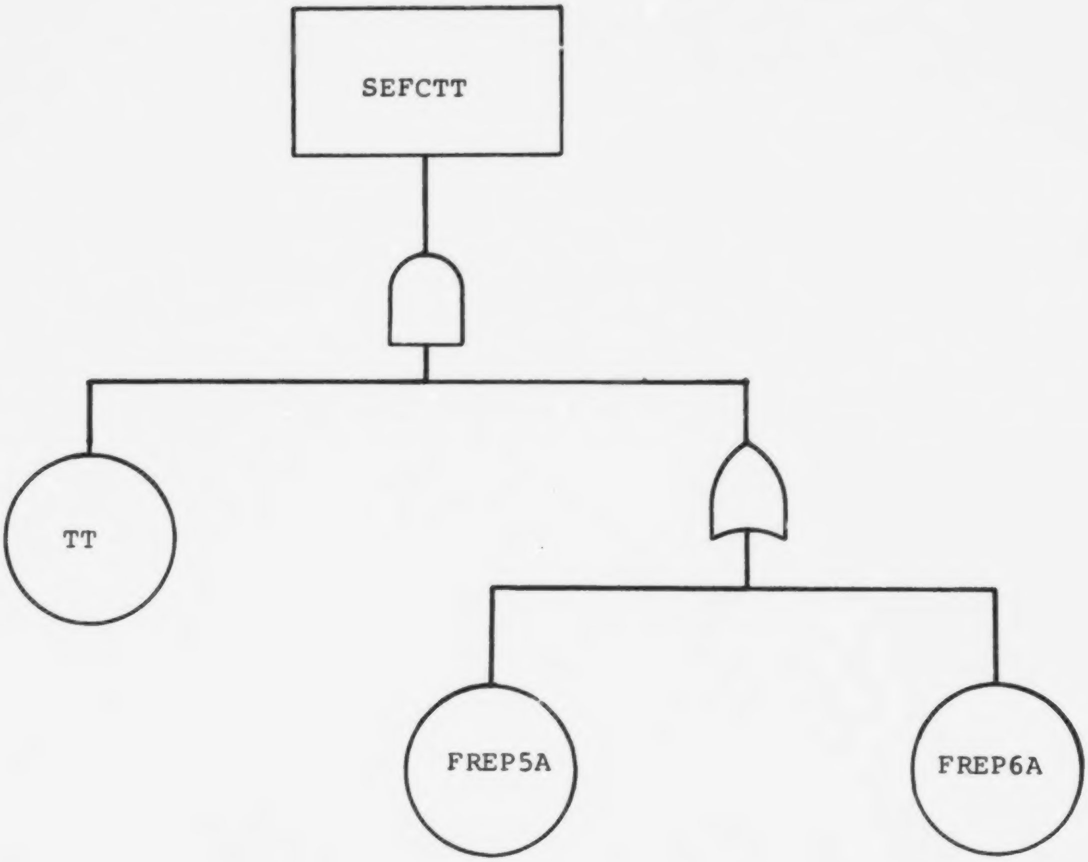
SEFCTT
TT
FREP5A
FREP6A

Figure A4-4. Sequence SEFCTT Fault Tree, Indian Point 2

Figure A4-5. Sequence SEHUR Fault Tree, Indian Point 2

A5. Indian Point 3 Dominant Sequences, System Level and Basic Event Cut Sets and Sequence Fault Trees.

Table A5-1

Indian Point 3

Description of Dominant Accident Sequences

Sequence*	Description
SLFHPRC	Failure of High Pressure Recirculation Cooling given a small LOCA.
SEFIRE	Specific fires in switchgear room and cable spreading room cause RCP seal LOCA and failure of Power Cables to the Safety Injection Pumps, Containment Spray Pumps, and Fan Coolers.
LLLPRC	Failure of Low Pressure Recirculation Cooling given large LOCA.
MLLPRC	Failure of Low Pressure Recirculation Cooling given Medium LOCA.

---

*The Indian Point 3 sequences correspond to sequences 1, 2, 3, and 4 of Table 8.3-10 of the Indian Point PSS.

Table A5-2

Indian Point 3

System Level Cut Sets of Significant Sequences

Sequence*	Cut Sets	Frequency (R-yr)$^{-1}$
1. SLFHPRC	S_2*OPERROR	6.9E-6
	S_2*OPERROR	9.4E-7
	S_2*RCIR	3.4E-10
	S_2*OPERROR*RCIR	3.1E-8
	S_2*RCIR	6.2E-5
	S_2*RCIR	1.2E-8
	S_2*RCIR	2.8E-7
	S_2*RCIR	1.1E-5
2. SEFIRE	FIRE	3.6E-5
	FIRE	2.4E-5
3. LLLPRC	A*OPERROR	9.3E-6
	A*OPERROR	1.0E-6
	A*OPERROR*RCIR	7.3E-8
	A*OPERROR*RCIR	2.5E-9
	A*RCIR	4.3E-12
	A*RCIR	6.4E-8
4. MLLPRC	A*OPERROR	9.3E-6
	A*OPERROR	1.0E-6
	A*OPERROR*RCIR	7.3E-8
	A*OPERROR*RCIR	2.5E-9
	A*RCIR	4.3E-12
	A*RCIR	6.4E-8

---

*The Indian Point 3 sequences correspond to sequences 1, 2, 3, and 4 of table 8.3-10 of the Indian Point PSS.

Table A5-3

Indian Point 3

Basic Event Cut Sets of Significant Sequences

Sequence*	Cut Sets	Frequency (R-yr)$^{-1}$
1. SLFHPRC	SLOCA*FIS13	6.9E-6
	SLOCA*SIPS	9.4E-7
	SLOCA*P60	3.4E-10
	SLOCA*RCP3132	3.1E-8
	SLOCA*SIP313233	6.2E-5
	SLOCA*MOV888883	1.2E-8
	SLOCA*SIP3	2.8E-7
	SLOCA*SIPMOV888	1.1E-5
2. SEFIRE	SWGEAR	3.6E-5
	CSRTE	2.4E-5
3. LLLPRC	LLOCA*FIS19	9.3E-6
	LLOCA*MOV746747	1.0E-6
	LLOCA*RP3132RHR	7.3E-8
	LLOCA*1802ABRHR	2.5E-9
	LLOCA*MSIP	4.3E-12
	LLOCA*HDWE	6.4E-8
4. MLLPRC	MLOCA*FIS19	9.3E-6
	MLOCA*MOV746747	1.0E-6
	MLOCA*RP3132RHR	7.3E-8
	MLOCA*1802ABRHR	2.5E-9
	MLOCA*MSIP	4.3E-12
	MLOCA*HDWE	6.4E-8

---

*The Indian Point 3 sequences correspond to sequences 1, 2, 3, and 4 of Table 8.3-10 of the Indian Point PSS.

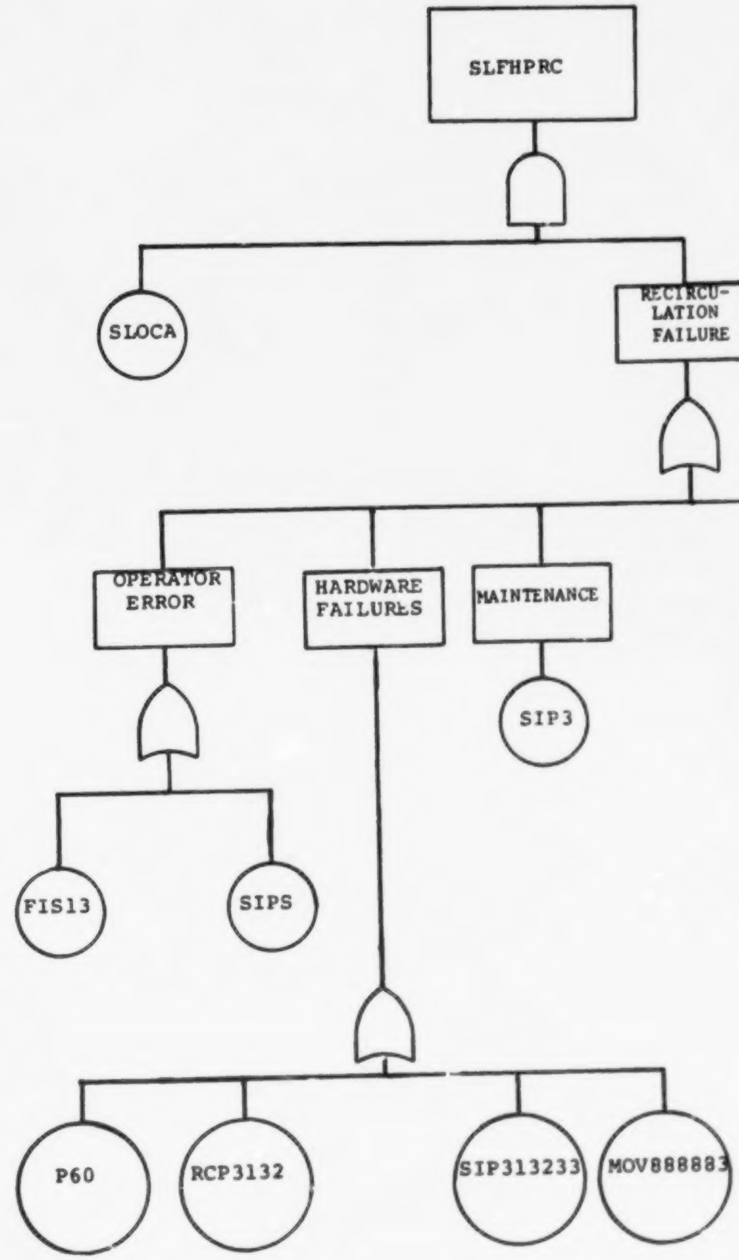
SLFHPRC
SLOCA
RECIRCU-LATION FAILURE
OPERATOR ERROR
HARDWARE FAILURES
MAINTENANCE
β FACTOR FAILURES
SIP3
SIPMOV888
FIS13
SIPS
P60
RCP3132
SIP313233
MOV888883

Figure A5-1. Sequence SLFHRPRC Fault Tree, Indian Point 3

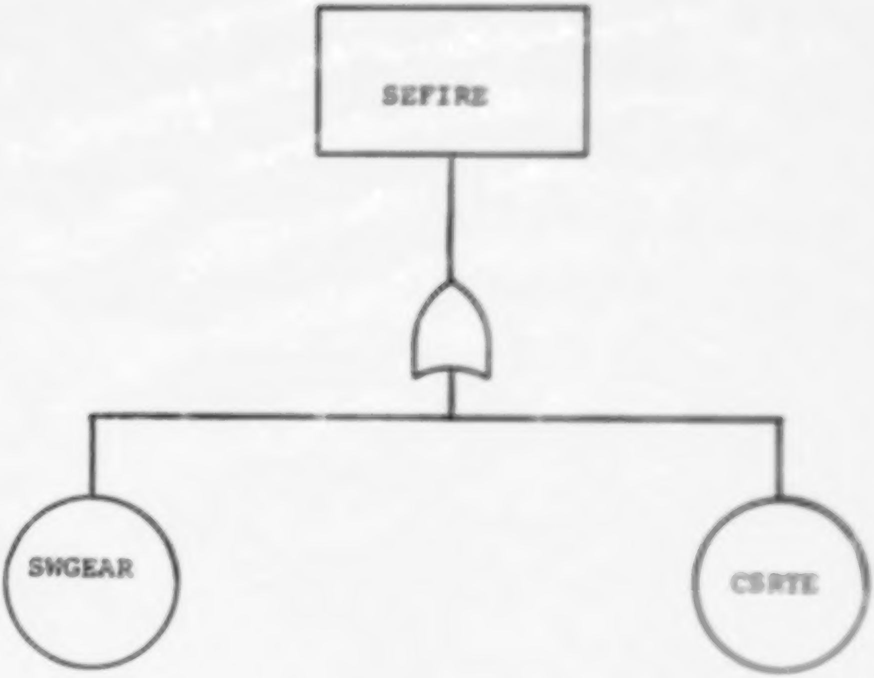
SEFIRE
SWGEAR
CSRTE

Figure A5-2. Sequence SEFIRE Fault Tree, Indian Point 3

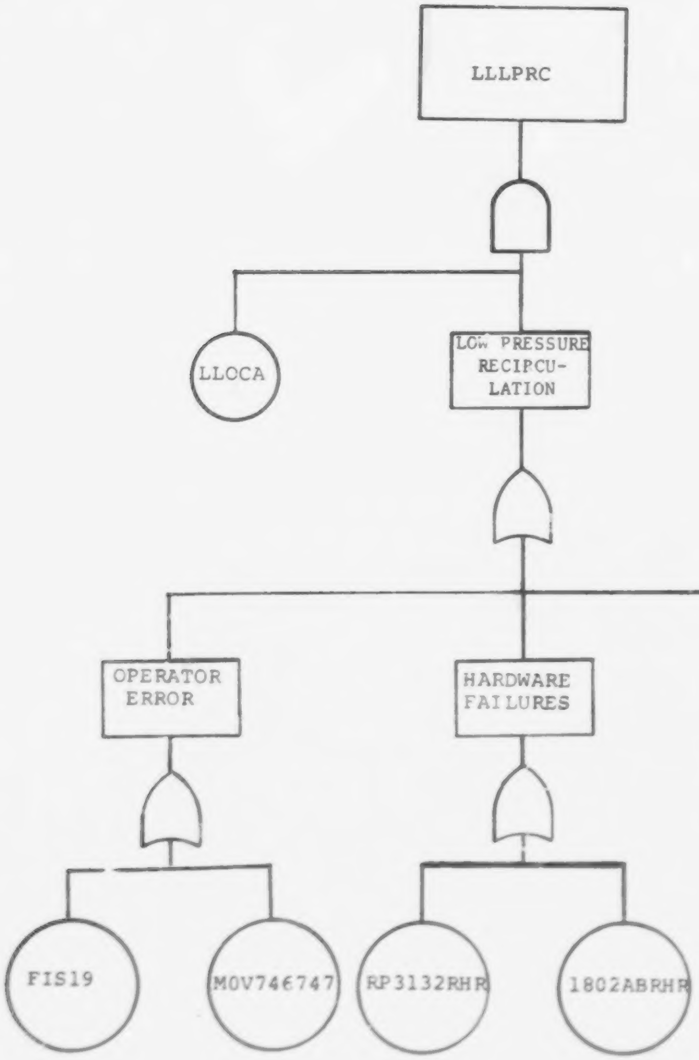
LLLPRC
LLOCA
LOW PRESSURE RECIRCU-LATION
OPERATOR ERROR
HARDWARE FAILURES
MAINTENANCE
β FACTORS FAILURES
FIS19
MOV746747
RP3132RHR
1802ABRHR
MSIP
HDWE

Figure A5-3. Sequence LLLPRC Fault Tree, Indian Point 3.

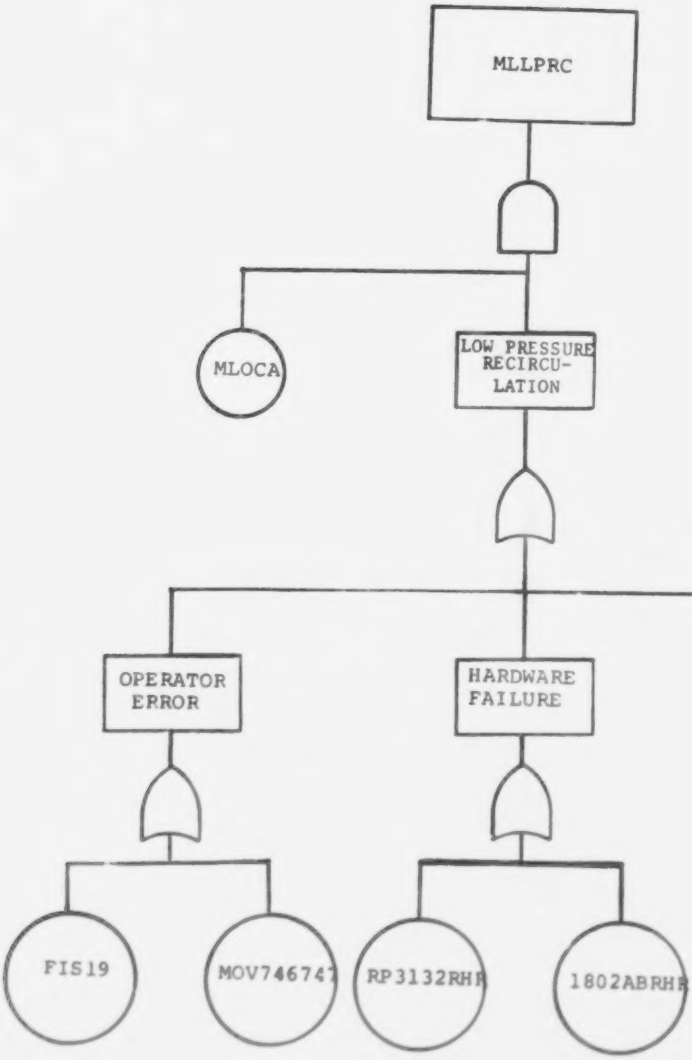
MLLPRC
MLOCA
LOW PRESSURE RECIRCU- LATION
OPERATOR ERROR
HARDWARE FAILURE
MAINTENANCE
β FACTORS FAILURES
MSIP
HDWE
FIS19
MOV746747
RP3132RHR
1802ABRHR

Figure A5-4. Sequence MLLPRC Fault Trees, Indian Point 3

A6. Crystal River-3 Dominant Sequences, System Level and Basic Event Cut Sets.

Table A6-1

Crystal River -3

Description of Dominant Accident Sequences

Sequence	Description
B_4S_{23}	Small-small LOCA followed by failure of Emergency Coolant Injection and Reactor Building Spray Injection.
B_4S_2	Small-small LOCA followed by failure of High Pressure Recirculation.
$T_{2A}T_{10}$	Loss of offsite power transient followed by failure of Emergency Feedwater, Primary Systems Make-up, Containment Pressure Reduction and Post Accident Radioactivity Removal.

Table A6-2

Crystal River-3

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
B_4S_{23}	S_2*OPERROR	6.5E-5
B_4S_2	S_2*OPERROR	1.0E-4
	S_2*OPERROR	6.5E-5
$T_{2A}T_{10}$	T_1*EACPOWER*DCPOWER	2.3E-5
	T_1*EACPOWER*EFWS	9.6E-6
	T_1*EACPOWER*DCPOWER	5.9E-6

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Table A6-3

Crystal River-3

Basic Event Cut Sets of Significant Sequences*

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
B_4S_{23}	B_4*LO17	6.5E-5
B_4S_{23}	B_4*Hx02 B_4*LO6	1.0E-4
$T_{2A}T_{10}$	T_{2A}*A2*A3*DCB	2.3E-5
	T_{2A}*A2*A3*A5*E1	9.6E-6
	T_{2A}*A2*DC13*AM1	5.9E-6

---

*Crystal River Cut sets are taken directly from the Crystal River IREP study.

A7. ANO-1 Dominant Sequences, System Level and Basic Event Cut Sets.

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Table A7-1

ANO-1

Description of Dominant Accident Sequences

Sequence	Description
B(1.2)D,C	Failure of the High Pressure Injection System (HPIS) and Reactor Building Spray Injection System (RBSI) given a small LOCA ($.38" < D \leq 1.2"$) in the Reactor Coolant System (RCS).
B(1.2)D_1	Failure of the HPIS following a B(1.2) sized small LOCA.
T(A3)LQ-E_3	Failure of the Engineered Safeguards power bus A3 with concomitant failure of the power conversion system (PCS) followed by failure of Emergency Feedwater System (EFWS), failure of a safety/relief valve to reclose, and failure of 2 out of 3 High Pressure Injection Pumps.
T(LOP)LD_1YC	Loss of offsite power with failure of PCS, followed by failure of EFWS, HPIS, RBSI, and Reactor Building Cooling System (RBCS).
T(DO1)LD_1YC	This sequence is the same as T(LOP)LD_1YC, except that it is initiated by a failure of the Engineered Safeguards power bus DO1.
T(DO2)LD_1YC	This sequence is the same as TC(DO1)LD,YC, expect that it is initiated by a failure of the Engineered Safeguards power bus DO2.
T(FIA)KD_1	Failure of Reactor Protection System (RPS) given a transient event, followed by failure of HPIS.
T(DO1)LD_1	Failure of Engineered Safeguards power bus DO1 with failure of PCS, followed by failure of EFWS and HPIS.
T(DO1)L1-D_3	Failure of the Engineered Safeguards power bus DO1 and the PCS, followed by failure of the EFWS, a safety/relief valve to reseal, and failure of 2 out of 3 HPIS pumps.

Table A7-2

ANO-1

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
B(1.2)D_1	S_3*HPIS*ECS	6.4E-8
	S_3*HPIS*SWS	2.8E-8
	S_3*HPIS*SWS	2.8E-8
	S_3*HPIS	5.3E-7
	S_3*OPERROR	2.0E-6
	S_3*HPIS*SWS	7.0E-8
	S_3*HPIS*ECS	1.4E-8
	S_3*HPIS*ACPOWER	6.2E-9
	S_3*HPIS*ACPOWER	1.5E-8
B(1.2)D_1C	S_3*HPIS*RBSI*ECS	3.8E-8
	S_3*HPIS*RBSI*ECS	3.8E-8
	S_3*HPIS*RBSI	2.0E-6
	S_3*HPIS*RBSI*SWS	1.6E-8
	S_3*HPIS*RBSI*SWS	1.6E-8
	S_3*HPIS*RBSI	1.4E-6
	S_3*HPIS*RBSI*SWS	4.2E-8
	S_3*ECS*HPIS*RBSI	8.2E-9
	S_3*HPIS*RBSI*SWS	3.2E-9
	S_3*ECS*HPIS*RBSI	8.2E-9
	S_3*RBSI*HPIS*ECS	2.2E-9
T(A3)LQ-D_3	T(A3)*EFIC*Q	1.6E-7
	T(A3)*EFWS*Q	2.8E-6
	T(A3)*EFWS*Q	2.2E-7
	T(A3)*EFWS*Q	2.6E-8
	T(A3)*EFIC*Q	4.2E-9
	T(A3)*EFWS*EFWS*Q	4.0E-9
	T(A3)*EFWS*EFIC*Q	4.0E-9
T(LOP)LD_1YC	T_1*DCPOWER	8.3E-6
	T_1*EACPOWER*EFIC	8.0E-8
	T_1*EACPOWER*EFWS	5.0E-7
	T_1*ECS*EACPOWER*EFIC	1.1E-8
	T_1*ECS*EACPOWER*EFIC	9.9E-9
	T_1*SWS*EACPOWER*EFIC	2.1E-8
	T_1*SWS*EACPOWER*EFIC	2.1E-8
	T_1*DCPOWER	1.4E-7
	T_1*SWS*EACPOWER*EFWS	3.8E-8
	T_1*SWS*EACPOWER*EFWS	3.8E-8

Table A7-2 (continued)

ANO-1

Sequence	Cut Sets	Frequency $(\text{R-yr})^{-1}$
T(D01)LD$_1$YC	T(D01)*SWS*EFWS	5.0E-8
	T(D01)*SWS*EFIC	4.5E-8
	T(D01)*SWS*EFIC	4.0E-8
	T(D01)*SWS*EFWS	3.4E-8
	T(D01)*SWS*EFIC	2.2E-8
	T(D01)*SWS*EFIC	2.2E-8
	T(D01)*DCPOWER	1.8E-6
	T(D01)*SWS*EFWS	1.7E-8
	T(D01)*SWS*EFIC	1.6E-8
	T(D01)*SWS*EFWS	5.4E-8
	T(D01)*ECS*EFWS	1.1E-8
	T(D01)*SWS*EFIC	5.0E-8
	T(D01)*SWS*EFIC	9.9E-9
	T(D01)*SWS*EFIC	4.3E-8
	T(D01)*ECS*EFIC	8.5E-9
	T(D01)*SWS*EFWS	3.6E-8
	T(D01)*ECS*EFWS	7.4E-9
	T(D01)*SWS*EFIC	2.5E-8
	T(D01)*ECS*EFIC	4.9E-7
	T(D01)*SWS*EFIC	2.2E-8
	T(D01)*ECS*EFIC	4.1E-9
	T(D01)*SWS*EFWS	1.8E-8
	T(D01)*ECS*EFWS	3.8E-9
	T(D01)*SWS*EFIC	1.8E-8
	T(D01)*ECS*EFIC	3.6E-9
	T(D01)*ACPOWER*EFWS	2.3E-8
	T(D01)*ACPOWER*EFIC	4.3E-9
	T(D01)*ACPOWER*EFIC	3.8E-9
	T(D01)*ACPOWER*EFWS	1.4E-8
	T(D01)*ACPOWER*EFWS	1.3E-8
	T(D01)*ACPOWER*EFIC	2.3E-9
	T(D01)*ACPOWER*EFIC	2.2E-9
	T(D01)*ACPOWER*EFIC	2.2E-9
	T(D01)*ACPOWER*EFIC	1.8E-9
	T(D01)*ACPOWER*EFWS	7.6E-9
	T(D01)*ACPOWER*EFWS	3.2E-8
	T(D01)*ACPOWER*EFIC	1.4E-8
T(DO2)LD$_1$YC	T(D02)*DCPOWER	1.8E-6
	T(D02)*SWS*EFWS	1.2E-9
	T(D02)*ACPOWER*EFWS	2.3E-8
	T(D02)*ACPOWER*EFIC	4.3E-9
	T(D02)*ACPOWER*EFIC	3.8E-9
	T(D02)*ACPOWER*EFWS	1.4E-8
	T(D02)*SWS*EFWS	5.9E-10
	T(D02)*SWS*EFWS*EFIC	5.4E-10

Table A7-2 (continued)

ANO-1

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T(D02)LD$_1$YC (continued)	T(D02)*SWS*EFWS	5.4E-10
	T(D02)*SWS*EFWS	5.4E-10
	T(D02)*ACPOWER*EFWS	1.3E-8
	T(D02)*SWS*EFIC	5.0E-10
	T(D02)*ACPOWER*EFIC	2.3E-9
	T(D02)*ACPOWER*EFIC	2.2E-9
	T(D02)*ACPOWER*EFWS	7.6E-9
T(FIA)KD$_1$	T$_3$*RPS*OPERROR	7.1E-7
	T$_3$*RPS*OPERROR	7.1E-7
	T$_3$*RPS*OPERROR	7.1E-7
	T$_3$*RPS*OPERROR	7.1E-7
T(D01)LD$_1$	T(D01)*HPIS*EFWS	7.0E-7
	T(D01)*HPIS*EFIC	1.4E-7
	T(D01)*HPIS*EFIC	1.2E-7
	T(D01)*HPIS*EFWS	4.5E-7
	T(D01)*OPERROR*EFWS	2.7E-8
	T(D01)*OPERROR*EFIC	5.4E-9
	T(D01)*OPERROR*EFIC	4.5E-9
	T(D01)*HPIS*EFIC	6.8E-8
	T(D01)*OPERROR*EFWS	1.6E-8
	T(D01)*HPIS*EFIC	5.8E-8
	T(D01)*HPIS*EFWS	2.3E-7
	T(D01)*HPIS*EFIC	4.9E-8
	T(D01)*OPERROR*EFWS	4.0E-8
T(D01)LQ-D$_3$	T(D01)*EFWS*Q	1.1E-6
	T(D01)*EFIC*Q	2.0E-7
	T(D01)*EFIC*Q	1.7E-7
	T(D01)*EFIC*Q	6.4E-7
	T(D01)*EFIC*Q	9.8E-8
	T(D01)*EFIC*Q	8.2E-8
	T(D01)*EFWS*Q	1.5E-6
	T(D01)*EFIC*Q	7.2E-8
	T(D01)*EFWS*Q	1.1E-7
	T(D01)*EFWS*Q	3.6E-8
	T(D01)*EFIC*Q	1.0E-9
	T(D01)*EFIC*Q	4.6E-10

Table A7-3

ANO-1

Basic Event Cut sets of Significant Sequences*

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
B(1.2)D_1	B(1.2)*HPI-H14*SWS-VCH4B	6.4E-8
	B(1.2)*HPI-H14*SWS-S14	2.8E-8
	B(1.2)*HPI-H14*SWS-S5	2.8E-8
	B(1.2)*A1*HPI-H14	5.3E-7
	B(1.2)*A2	2.0E-6
	B(1.2)*HPI-H14*SWS-S2	7.0E-8
	B(1.2)*HPI-H14*A3	1.4E-8
	B(1.2)*HPI-H14*AC-B5	6.2E-9
	B(1.2)*HPI-H14*AC-A3	1.5E-8
B(1.2)D_1C	B(1.2)*A4*SWS-VCH4B	3.8E-8
	B(1.2)*A1*SWS-VCH4A	3.8E-8
	B(1.2)*LPI-L25	2.0E-6
	B(1.2)*A4*SWS-S14	1.6E-8
	B(1.2)*A4*SWS-S5	1.6E-8
	B(1.2)*A1*A4	1.4E-6
	B(1.2)*A4*SWS-S2	4.2E-8
	B(1.2)*A4*A3	8.2E-9
	B(1.2)*A1*SWS-S1	3.2E-7
	B(1.2)*A1*ECS-ROOM99	8.2E-9
	B(1.2)*A5*HPI-H14*SWS-VCH4B	2.2E-9
T(A3)LQ-D_3	T(A3)*EFC-D1D2*Q	1.6E-7
	T(A3)*EFW-E8*Q	2.8E-6
	T(A3)*EFW-E22*Q	2.2E-7
	T(A3)*EFW-E4*EFW-E5*Q	2.6E-8
	T(A3)*EFC-ABCD4*EFC-ACBD4*Q	4.2E-9
	T(A3)*EFW-E5*EFC-VCD2*Q	4.0E-9
	T(A3)*EFW-E4*EFC-VCD1*Q	4.0E-9
T(LOP)LD_1YC	T(LOP)*EBAT-CM	8.3E-6
	T(LOP)*AC-DG1*AC-DG2*EFC-D1D2	8.0E-8
	T(LOP)*AC-DG1*AC-DG2*EFW-E8	5.1E-7
	T(LOP)*SWS-VCH4A*AC-DG1*EFC-D1D2	1.1E-8
	T(LOP)*SWS-VCH4A*AC-DG1*EFW-E8	9.9E-9
	T(LOP)*SWS-S62*AC-DG2*EFC-D1D2	2.1E-8
	T(LOP)*SWS-S63*AC-DG1*EFC-D1D2	2.1E-8
	T(LOP)*DC-D06*DC-D07	1.4E-7
	T(LOP)*SWS-S62*AC-DG2*EFW-E8	3.8E-8
	T(LOP)*SWS-S63*AC-DG1*EFW-E8	3.8E-8

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*ANO cut sets are taken directly from the ANO-1 IREP study.

Table A7-3 (Continued)

ANO-1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T(D01)LD$_1$YC	T(D01)*SWS-VCH4A*EFW-E4	5.0E-8
	T(D01)*SWS-VCH4A*EFC-ACBD4	4.5E-8
	T(D01)*SWS-VCH4A*EFC-VCD2	4.0E-8
	T(D01)*SWS-VCH4A*EFW-E29	3.4E-8
	T(D01)*SWS-VCH4A*EFC-BB7B1	2.2E-8
	T(D01)*SWS-VCH4A*EFC-D1D2	2.2E-8
	T(D01)*DC-D02	1.8E-6
	T(D01)*SWS-VCH4A*EFW-E8	1.7E-8
	T(D01)*SWS-VCH4A*EFC-CSY2	1.6E-8
	T(D01)*SWS-S1*EFW-E4	5.4E-8
	T(D01)*ECS-ROOM99*EFW-E4	1.1E-8
	T(D01)*SWS-S1*EFC-ACBD4	5.0E-8
	T(D01)*SWS-ROOM99*EFC-ACBD4	9.9E-9
	T(D01)*SWS-S1*EFC-VCD2	4.3E-8
	T(D01)*ECS-ROOM99*EFC-VCD2	8.5E-9
	T(D01)*SWS-S1*EFW-E29	3.6E-8
	T(D01)*ECS-ROOM99*EFW-E29	7.4E-9
	T(D01)*SWS-S1*EFC-BB7B1	2.5E-8
	T(D01)*ECS-ROOM99*EFC-BB7B1	4.9E-7
	T(D01)*SWS-S1*EFC-D1D2	2.2E-8
	T(D01)*ECS-ROOM99*EFC-D1D2	4.1E-9
	T(D01)*SWS-S1*EFW-E8	1.8E-8
	T(D01)*ECS-ROOM99*EFW-E8	3.8E-9
	T(D01)*SWS-S1*EFC-CSY2	1.8E-8
	T(D01)*ECS-ROOM99*EFC-CSY2	3.6E-9
	T(D01)*AC-B6*EFW-E4	2.3E-8
	T(D01)*AC-B6*EFC-ACBD4	4.3E-9
	T(D01)*AC-B6*EFC-VCD2	3.8E-9
	T(D01)*AC-B6*EFW-E29	1.4E-8
	T(D01)*AC-A4*EFW-E4	1.3E-8
	T(D01)*AC-A4*EFC-ACBD4	2.3E-9
	T(D01)*AC-B6*EFC-BB7B1	2.2E-9
	T(D01)*AC-A4*EFC-VCD2	2.2E-9
	T(D01)*AC-B6*EFC-D1D2	1.6E-9
	T(D01)*AC-A4*EFW-E29	7.6E-9
	T(D01)*AC-B6*EFW-E8	3.2E-8
	T(D01)*AC-B6*EFC-CSY2	1.4E-8

Table A7-3 (Continued)

ANO-1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T(D02)LD$_1$YC	T(D02)*DC-D01	1.8E-6
	T(D02)*SWS-VCA4B*EFW-E22	1.2E-9
	T(D02)*AC-B5*EFW-E5	2.3E-8
	T(D02)*AC-B5*EFC-ABCD4	4.3E-9
	T(D02)*AC-B5*EFC-VCD1	3.8E-9
	T(D02)*AC-B5*EFW-E28	1.4E-8
	T(D02)*SWS-VCH4B*EFW-E5*EFW-E3	5.9E-10
	T(D02)*SWS-VCH4B*EFW-E3*EFC-ABCD4	5.4E-10
	T(D02)*SWS-S14*EFW-E22	5.4E-10
	T(D02)*SWS-S5*EFW-E22	5.4E-10
	T(D02)*AC-A3*EFW-E5	1.3E-8
	T(D02)*SWS-VCH4B*EFC-ABCD4*EFC-ACBD4	5.0E-10
	T(D02)*AC-A3*EFC-ABCD4	2.3E-9
	T(D02)*AC-A3*EFC-VCD1	2.2E-9
	T(D02)*AC-A3*EFW-E28	7.6E-9
T(FIA)KD$_1$	T(FIA)*A8*A9*PUMP	7.1E-7
	T(FIA)*A8*A10*PUMP	7.1E-7
	T(FIA)*A11*A12*PUMP	7.1E-7
	T(FIA)*A11*A13*PUMP	7.1E-7
T(D01)LD$_1$	T(D01)*HPI-H14*EFW-E4	7.0E-7
	T(D01)*HPI-H14*EFC-ACBD4	1.4E-7
	T(D01)*HPI-H14*EFC-VCD2	1.2E-7
	T(D01)*HPI-H14*EFW-E29	4.5E-7
	T(D01)*HPI-PUMP*EFW-E4	2.7E-8
	T(D01)*HPI-PUMP*EFC-ACBD4	5.4E-9
	T(D01)*HPI-PUMP*EFC-VCD2	4.5E-9
	T(D01)*HPI-H14*EFC-BB7B1	6.8E-8
	T(D01)*HPI-PUMP*EFW-E29	1.6E-8
	T(D01)*HPI-H14*EFC-D1D2	5.8E-8
	T(D01)*HPI-H14*EFW-E8	2.3E-7
	T(D01)*HPI-H14*EFC-C2Y2	4.9E-8
	T(D01)*HPI-PUMP*EFW-E8	4.0E-8
T(D01)LQ-D$_3$	T(D01)*EFW-E4*Q	1.1E-6
	T(D01)*EFC-ACBD4*Q	2.0E-7
	T(D01)*EFC-VCD2*Q	1.7E-7
	T(D01)*EFW-E29*Q	6.4E-7
	T(D01)*EFC-BB7B1*Q	9.8E-8

Table A7-3 (Continued)

ANO-1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T(D01)LQ-D_3 (continued)	T(D01)*EFC-D1D2*Q	8.2E-8
	T(D01)*EFW-E8*Q	1.5E-6
	T(D01)*EFC-CSY2*Q	7.2E-8
	T(D01)*EFW-E22*Q	1.1E-7
	T(D01)*EFW-E2*Q	3.6E-8
	T(D01)*EFC-VE4*EFC-ABCD4*Q	1.0E-9
	T(D01)*EFC-VE4*EFC-VE2*Q	4.6E-10

A8. Oconee Dominant Sequences, System Level and Basic Event Cut Sets.

Table A8-1

Oconee

Description of Dominant Accident Sequences

Sequence	Description
T_2MQ-H	Loss of Power Conversion System (PCS), followed by failure of a safety/relief value to reseal, and failure of the Emergency Coolant Recirculation System (ECRS).
S_3H	Rupture of the Reactor Cooling System (RCS) of size $S_3(0 < D < 4")$ followed by failure of ECRS.
S_1D	A S_1-sized LOCA $(10" < D \leq 13.5")$ followed by failure of the Emergency Coolant Injection System (ECIS).
T_2MQ-FH	Loss of the PCS followed by failure of one safety/relief valve to release, failure of the Containment Spray Recirculation System (CSRS), and the ECRS.
S_3FH	S_3-sized LOCA followed by failure of CSRS and ECRS.
T_2MLUO	Loss of the PCS followed by failure to restore PCS, failure of the EFWS, HPIS, and the Reactor Building Cooling System (RBCS).
T_2MKU	Loss of PCS followed by failure of the Reactor Protection System (RPS) and HPIS.
T_1MLUO	This sequence is the same as T_2MLUO, except that loss of PCS is initiated by loss of offsite power.
V	Interfacing System LOCA failure of LPIS check valves.

Table A8-2

Oconee

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_2MQ-H	T_2*Q*OPERROR	4.5E-6
	T_2*Q*OPERROR	4.5E-6
	T_2*Q*LPRS	7.4E-7
	T_2*Q*LPRS	3.2E-7
	T_2*Q*LPRS	3.2E-7
S_3H	S_3*OPERROR	3.9E-6
	S_3*OPERROR	3.9E-6
	S_3*LPRS	6.4E-7
	S_3*LPRS	2.7E-7
	S_3*LPRS	2.7E-7
S_1D	S_1*LPIS	2.3E-6
	S_1*LPIS	2.3E-6
	S_1*ESAD	5.0E-7
	S_1*ESAD	5.0E-7
	S_1*LPIS	3.3E-7
	S_1*LPIS	3.3E-7
	S_1*OPERROR	3.0E-7
T_2MQ-FH	T_2*Q*OPERROR	4.5E-6
	T_2*Q*LPRS*CSRS	4.1E-8
	T_2*Q*LPRS*CSRS	4.1E-8
	T_2*Q*LPRS*CSRS	1.3E-7
S_3FH	S_3*OPERROR	3.9E-6
	S_3*LPRS*CSRS	1.1E-7
	S_3*LPRS*CSRS	3.5E-8
	S_3*LPRS*CSRS	3.5E-8
T_2MLUO	T_2*PCS*LPSW	6.0E-6
	T_2*PCS*LPSW*ESAD	2.1E-6
T_2KMU	T_2*RPS*OPERROR	7.8E-6
T_1MLUO	T_1*LPSW	4.0E-6
	T_1*LPSW*ESAD	1.4E-6
V	V	4.0E-6

Table A8-3

Oconee

Basic Event Cut Sets of Significant Sequences*

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_2MQ-H	T_2*Q*HPRSCM	4.5E-6
	T_2*Q*LPISCM	4.5E-6
	T_2*Q*D	7.4E-7
	T_2*Q*E	3.2E-7
	T_2*Q*D	3.2E-7
S_3H	S_3*HPRSCM	3.9E-6
	S_3*LPISCM	3.9E-6
	S_3*D*E	6.4E-7
	S_3*E*W	2.7E-7
	S_3*D*X	2.7E-7
S_1D	S_1*D	2.3E-6
	S_1*E	2.3E-6
	S_1*CH3	5.0E-7
	S_1*CH4	5.0E-7
	S_1*C	3.3E-7
	S_1*B	3.3E-7
	S_1*LPISCM	3.0E-7
T_2MQ-FH	T_2*Q*WXCM	4.5E-6
	T_2*Q*B*W	4.1E-8
	T_2*Q*C*X	4.1E-8
	T_2*Q*W*X	1.3E-7
S_3FH	S_3*WXCM	3.9E-6
	S_3*W*X	1.1E-7
	S_3*B*W	3.5E-8
	S_3*C*X	3.5E-8
T_2MLUO	T_2*PCSNR*F1*G1	6.0E-6
	T_2*PCSNR*F1*CH4	2.1E-6
T_2KMU	T_2*K*HPMANK	7.8E-6
V	V	4.0E-6

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*Oconee cut set are taken directly from the Oconee RSSMAP Study.

A9. Calvert Cliffs Dominant Sequences, System Level and Basic Event Cut Sets.

Table A9-1

Calvert Cliffs

Description of Dominant Accident Sequences

Sequence	Description
T_2ML	Loss of Power Conversion System (PCS) followed by failure to restore PCS and failure of Auxillary Feedwater System (AFWS).
T_1ML	Loss of offsite power, PCS, and AFWS.
T_3ML	Transient event with PCS initially available, followed by loss of PCS and AFWS.
$T_1MLO(O)$	Loss of offsite power with concomitant failure of PCS, AFWS, Containment Air Recirculation and Cooling System, and Containment Spray Injection System.

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Table A9-2

Calvert Cliffs

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_2ML	T_2*AFWS*OPERROR	9.0E-4
T_1ML	T_1*AFWS*OPERROR	6.0E-4
	T_1*RECOVERY*EACPOWER	2.6E-5
	T_1*RECOVERY*EACPOWER*SALTWS	1.7E-5
T_3ML	T_3*PCS*AFWS*OPERROR	1.2E-4
T_1MLO(O)	T_1*DCPOWER	8.0E-5
	T_1*EACPOWER*LOPNR*RECOVERY	5.4E-6
	T_1*SALTWS*LOPNR*RECOVERY* EACPOWER	3.6E-6
	T_1*RECOVERY*LOPNR*EACPOWER	3.6E-6
	T_1*RECOVERY*LOPNR*EACPOWER* SALTWS	1.8E-6

Table A9-3

Calvert Cliffs

Basic Event Cut Sets of Significant Sequences*

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_2ML	T_2*CONST1*PCSNR	9.0E-4
T_1ML	T_1*CONST1	6.0E-4
	T_1*LOACRES*D21ST*D12ST	2.6E-5
	T_1*LOACRES*D12SR*F2	1.7E-5
T_3ML	T_3*PCST3*CONST1	1.2E-4
T_1MLO(O)	T_1*BATCM	8.0E-5
	T_1*D21ST*D12ST*LOPNR8HR*DGNR8HR	5.4E-6
	T_1*D21ST*F2*LOPNR8HR*DGNR8HR	3.6E-6
	T_1*LOACRES*LOPNR1*D21ST*D12ST	2.6E-6
	T_1*LOACRES*LOPNRL*D12ST*F2	1.8E-6

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*Calvert Cliffs cut sets are taken directly from the Calvert Cliffs RSSMAP PRA.

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A10. Browns Ferry-1 Dominant Sequences, System Level and Basic Event Cut Sets.

Table A10-1

Browns Ferry-1

Description of Dominant Accident Sequences

Sequence	Description
$T_U R_B R_A$	Loss of Power Conversion Systems (PCS) followed by loss of the Residual Heat Removal System (RHRS).
$T_U B$	Loss of PCS followed by failure of the Reactor Protection System (RPS).
$T_P R_B R_A$	Loss of offsite power followed by failure of the (RHRS).

Table A10-2

Browns Ferry-1

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
$T_UR_BR_A$	T_3*RHRS*OPERROR	8.0E-8
	T_3*RHRS*OPERROR	8.0E-8
	T_3*RHRS*OPERROR	5.6E-8
	T_3*RHRS*OPERROR	5.6E-8
	T_3*OPERROR	1.7E-8
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	5.4E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW	3.6E-9
	T_3*RHRS*RHRSW*OPERROR	1.1E-9

Table A10-2 (continued)

Browns Ferry-1

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_UB	T_3*RPS	5.1E-5
$T_PR_BR_A$	T_1*RHRS*OPERROR	1.4E-7
	T_1*RHRS*OPERROR	1.4E-7
	T_1*RHRS*OPERROR	9.9E-8
	T_1*RHRS*OPERROR	9.9E-8
	T_1*OPERROR	3.0E-8
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW*OPERROR	2.0E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW*OPERROR	2.0E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW*OPERROR	2.0E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW*OPERROR	2.0E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	9.4E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW	6.6E-9
	T_1*RHRS*RHRSW*OPERROR	2.0E-9

Table A10-2 (continued)

Browns Ferry-1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
$T_PR_BR_A$ (Continued)	T_1*RHRS*RHRSW*OPERROR	2.0E-9
	T_1*EACPOWER	2.6E-5
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER	8.1E-9
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10
	T_1*EACPOWER*EECW	8.7E-10

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Table A10-3

Browns Ferry-1

Basic Event Cut Sets of Significant Sequences

Sequence	Cut Sets*	Frequency $(R\text{-}yr)^{-1}$
$T_U R_B R_A$	T_U*RR3A112*RRB0001D	8.0E-8
	T_U*RR3A122*RRB0001D	8.0E-8
	T_U*RCK048UG*RRB0001D	5.6E-8
	T_U*RCK0K7UG*RRB0001D	5.6E-8
	T_U*RRA0001D*RRB0001D	1.7E-8
	T_U*RR3A112*SUPRA*RCK0302G	5.4E-9
	T_U*RR3A122*SUPRA*RCK0302G	5.4E-9
	T_U*RCK048UG*SUPRA*RCK0302G	3.6E-9
	T_U*RCK047UG*SUPRA*RCK0302G	3.6E-9
	T_U*RRA0001D*SUPRA*RCK0302G	1.1E-9
	T_U*RR3A112*SUPRA*RCK0712G	5.4E-9
	T_U*RR3A122*SUPRA*RCK0712G	5.4E-9
	T_U*RCK048UG*SUPRA*RCK0712G	3.6E-9
	T_U*RCK047UG*SUPRA*RCK0712G	3.6E-9
	T_U*RRA0001D*SUPRA*RCK0712G	1.1E-9
	T_U*RR3A112*RCK0571G*SUPRB	5.4E-9
	T_U*RR3A122*RCK0571G*SUPRB	5.4E-9
	T_U*RCK048UG*RCK0571G*SUPRB	3.6E-9
	T_U*RCK047UG*RCK0571G*SUPRB	3.6E-9
	T_U*RRA0001D*RCK0571G*SUPRB	1.1E-9
	T_U*RR3A112*SUPRC*RCK0712G	5.4E-9
	T_U*RR3A122*SUPRC*RCK0712G	5.4E-9
	T_U*RCK048UG*SUPRC*RCK0712G	3.6E-9
	T_U*RCK047UG*SUPRC*RCK0712G	3.6E-9
	T_U*RRA0001D*SUPRC*RCK0712G	1.1E-9
	T_U*RR3A112*RCK0571G*SUPRD	5.5E-9
	T_U*RR3A122*RCK0571G*SUPRD	5.5E-9
	T_U*RCK048UG*RCK0571G*SUPRD	3.6E-9
	T_U*RCK047UG*RCK0571G*SUPRD	3.6E-9
	T_U*RRA0001D*RCK05716*SUPRD	1.1E-9
	T_U*RR3A112*SUPRC*RCK0302G	5.4E-9
	T_U*RR3A122*SUPRC*RCK0302G	5.4E-9
	T_U*RCK048UG*SUPRC*RCK0302G	3.6E-9
	T_U*RCK047UG*SUPRC*RCK0302G	3.6E-9

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*Browns Ferry cut sets are generated by combining the system cut set in Appendix B of the Browns Ferry IREP study according to the logic of the various sequences.

Table A10-3 (continued)

Browns Ferry-1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
$T_U R_B R_A$	T_U*RRA0001D*SUPRC*RCK0302G	1.1E-9
	T_U*RR3A112*RCK0071G*SUPRB	5.4E-9
	T_U*RR3A122*RCK0071G*SUPRB	5.4E-9
	T_U*RCK048UG*RCK0071G*SUPRB	3.6E-9
	T_U*RCK047UG*RCK0071G*SUPRB	3.6E-9
	T_U*RRA0001D*RCK0071G*SUPRB	1.1E-9
	T_U*RR3A112*RCK0071G*SUPRD	5.4E-9
	T_U*RR3A122*RCK0071G*SUPRD	5.4E-9
	T_U*RCKO48UG*RCK0071G*SUPRD	3.6E-9
	T_U*RCKO48UG*RCK0071G*SUPRD	3.6E-9
	T_U*RRA0001D*RCK0071G*SUPRD	1.1E-9
$T_U B$	T_U*B	5.1E-5
$T_P R_B R_A$	T_P*RR3A112*RRB0001D	1.4E-7
	T_P*RR3A122*RRB0001D	1.4E-7
	T_P*RCK048UG*RRB0001D	9.9E-8
	T_P*RCK047UG*RRB0001D	9.9E-8
	T_P*RRA0001D*RRB0001D	3.0E-8
	T_P*RR3A112*SUPRA*RCK0302G	9.4E-9
	T_P*RR3A122*SUPRA*RCK0302G	9.4E-9
	T_P*RCK048UG*SUPRA*RCK0302G	6.6E-9
	T_P*RCK047UG*SUPRA*RCK0302G	6.6E-9
	T_P*RRA0001D*SUPRA*RCK0302G	2.0E-9
	T_P*RR3A112*SUPRA*RCK0712G	9.4E-9
	T_P*RR3A122*SUPRA*RCK0712G	9.4E-9
	T_P*RCK048UG*SUPRA*RCK0712G	6.6E-9
	T_P*RCK047UG*SUPRA*RCK0712G	6.6E-9
	T_P*RRA0001D*SUPRA*RCK0712G	2.0E-9
	T_P*RR3A112*RCK0571G*SUPRB	9.4E-9
	T_P*RR3A122*RCK051G*SUPRB	9.4E-9
	T_P*RCK048UG*RCK0571G*SUPRB	6.6E-9
	T_P*RCK047UG*RCK0571G*SUPRB	6.6E-9
	T_P*RRA0001D*RCK0571G*SUPRB	2.0E-9
	T_P*RR3A112*SUPRC*RCK0712G	9.4E-9
	T_P*RR3A122*SUPRC*RCK0712G	9.4E-9
	T_P*RCK048UG*SUPRC*RCK0712G	6.6E-9
	T_P*RCK047UG*SUPRC*RCK0712G	6.6E-9
	T_P*RRA0001D*SUPRC*RCK0712G	2.0E-9
	T_P*RR3A11D*RCK0571G*SUPRD	9.4E-9
	T_P*RR3A122*RCK0571G*SUPRD	9.4E-9
	T_P*RCK048UG*RCK0571G*SUPRD	6.6E-9
	T_P*RCK047UG*RCK0571G*SUPRD	6.6E-9
	T_P*RRA0001D*RCK0571G*SUPRD	2.0E-9
	T_P*RR3A112*SUPRC*RCK0302G	9.4E-9

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Table A10-3 (continued)

Browns Ferry-1

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
$T_P R_B R_A$	T_p*RR3A122*SUPRC*RCK0302G	9.4E-9
	T_p*RCKO48UG*SUPRC*RCK0302G	6.6E-9
	T_p*RCKO47UG*SUPRC*RCK0302G	6.6E-9
	T_p*RRA0001D*SUPRC*RCK0302G	2.0E-9
	T_p*RR3A112*RCK0071G*SUPRB	9.4E-9
	T_p*RR3A122*RCK0071G*SUPRB	9.4E-9
	T_p*RCKO48UG*RCK0071G*SUPRB	6.6E-9
	T_p*RCKO47UG*RCK0071G*SUPRB	6.6E-9
	T_p*RRA0001D*RCK0071G*SUPRB	2.0E-9
	T_p*RR3A112*RCK0071G*SUPRD	9.4E-9
	T_p*RR3A122*RCK0071G*SUPRD	9.4E-9
	T_p*RCKO48UG*RCK0071G*SUPRD	6.6E-9
	T_p*RCKO47UG*RCK0071G*SUPRD	6.6E-9
	T_p*RRA0001D*RCK0071G*SUPRD	2.0E-9
	T_p*DG3COMB	2.6E-5
	T_p*ADL003AR*ADL001CR	8.1E-9
	T_p*ADL003AR*ADL003BR	8.1E-9
	T_p*ADL003AR*ADL001DR	8.1E-9
	T_p*ADL003AR*ADL001CR	8.1E-9
	T_p*ADL001DR*ADL001B2	8.1E-9
	T_p*ADL001CR*ADL001DR	8.1E-9
	T_p*ADL001AR*ADL001BR	8.1E-9
	T_p*ADL003BR*ADL001DR	8.1E-9
	T_p*KCK008NG*ADL001CR	8.7E-10
	T_p*KCK001NG*ADL003BR	8.7E-10
	T_p*ADL003BR*KCK011SG	8.7E-10
	T_p*ADL003BR*KCK005SG	8.7E-10
	T_p*KCK008NG*ADL001DR	8.7E-10
	T_p*KCK001NG*ADL001DR	8.7E-10
	T_p*ADL003AR*KCK008SG	8.7E-10
	T_p*ADL003AR*KCK011SG	8.7E-10
	T_p*KCK001NG*ADL001CR	8.7E-10
	T_p*ADL001CR*KCK011SG	8.7E-10
	T_p*KCK005SG*ADL001DR	8.7E-10
	T_p*ADL003AR*KCK005SG	8.7E-10

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A11. Grand Gulf 1 Dominant Sequences, System Level and Basic Event Cutsets.

Table A11-1

Grand Gulf-1

Description of Dominant Accident Sequences

Sequences	Description
$T_{23}PQI$	Transient event requiring reactor shutdown followed by failure of safety/relief value to reseal the Power Conversion System (PCS), and failure of the Residual Heat Removal System (RHRS) to cool the suppression pool within 28 hours.
SI	Small LOCA followed by failure of the RHRS to remove decay heat from the suppression pool.
T_1QW	Loss of offsite power followed by failure of PCS, and the RHRS to remove decay heat from containment within 30 hours.
$T_{23}QW$	Transient event requiring reactor shutdown followed by failure of PCS and of the RHRS to remove heat from the containment within 30 hours.
$T_{23}C$	Failure of Reactor Protection System (RPS) given a transient event.

Table A11-2

Grand Gulf 1

System Level Cutsets for Significant Sequences

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_{23}PQI	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SPMS	
	T_{23}*Q*PCS*RECOVERY*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS*RHRS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*SSWS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*SPMS	
	T_{23}*Q*PCS*RECOVERY*RHRS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*RHRS*SSWS	
	T_{23}*Q*PCS*RECOVERY*RHRS	
	T_{23}*Q*PCS*RECOVERY*SPMS	
SI	S*SSWS	6.2E-7
	S*SSWS	4.2E-7
	S*SSWS	4.2E-7
	S*SSWS	3.2E-7
	S*SSWS	3.2E-7

Table A11-2 (continued)

Grand Gulf 1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
SI (continued)	S*SSWS*RHRS	2.8E-7
	S*SSWS*RHRS	2.8E-7
	S*SSWS	1.9E-7
	S*SSWS	1.6E-7
	S*SSWS	1.6E-7
	S*SSWS*RHRS	1.3E-7
	S*SSWS*RHRS	1.3E-7
	S*SSWS	1.3E-7
	S*SSWS*RHRS	1.1E-7
	S*SSWS*RHRS	1.1E-7
	S*SPMS	8.6E-8
	S*RHRS	8.8E-8
	S*SSWS	4.0E-8
	S*SSWS*RHRS	4.0E-8
	S*SSWS	4.0E-8
	S*SSWS*RHRS	4.0E-8
	S*SSWS	3.5E-8
	S*SSWS*RHRS	3.5E-8
	S*SSWS	3.5E-8
	S*SSWS*RHRS	3.5E-8
	S*SSWS*RHRS	2.7E-8
	S*SSWS*RHRS	2.7E-8
	S*SSWS*RHRS	2.5E-8
	S*SSWS	2.5E-8
	S*SSWS*RHRS	2.5E-8
	S*SSWS	2.5E-8
	S*SSWS*RHRS	2.4E-8
	S*SSWS*RHRS	2.4E-8
	S*SPMS	2.4E-8
	S*RHRS	2.4E-8
	S*RHRS*SSWS	2.4E-8
	S*RHRS*SSWS	2.4E-8
	S*RHRS	2.4E-8
	S*SPMS	2.4E-8
T_1QW	T_1*LOPNRE*LOPNRL*RECOVERY*EACPOWER	1.1E-6
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	6.4E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	6.4E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	4.5E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	4.5E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	3.7E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	2.6E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	2.6E-7

Table A11-2 (continued)

Grand Gulf 1

Sequence	Cut Sets	Frequency $(R\text{-}yr^{-1})$
T_1QW (continued)	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	2.4E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	2.4E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	1.9E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	1.4E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	1.4E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	1.0E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	1.0E-7
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS	5.4E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER *RCICS	3.7E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER *RCICS	3.7E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*EACPOWER	3.6E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*EACPOWER*SSWS	3.6E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*DCPOWER*EACPOWER	3.0E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*DCPOWER*EACPOWER	3.0E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*RCICS	2.5E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*RHRS	2.3E-8
	T_1*LOPNRE*LOPNRL*RECOVERY*SSWS*RHRS* EACPOWER	2.3E-8
$T_{23}QW$	T_{23}*PCS*RECOVERY*SSWS	3.2E-6
	T_{23}*PCS*RECOVERY*SSWS	1.9E-6
	T_{23}*PCS*RECOVERY*SSWS	1.9E-6
	T_{23}*PCS*RECOVERY*SSWS	9.4E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	3.0E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	2.6E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	2.6E-7
	T_{23}*PCS*RECOVERY*SSWS	2.6E-7
	T_{23}*PCS*RECOVERY*SSWS	2.6E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	1.9E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	1.9E-7
	T_{23}*PCS*RECOVERY*SSWS	1.9E-7
	T_{23}*PCS*RECOVERY*SSWS	1.9E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	1.8E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	1.8E-7
	T_{23}*PCS*RECOVERY*SSWS	1.7E-7
	T_{23}*PCS*RECOVERY*SSWS	1.7E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	1.5E-7
	T_{23}*PCS*RECOVERY*SSWS*RCICS	1.5E-7
$T_{23}C$	T_{23}*RPS	5.4E-6

Table A11-3

Grand Gulf 1

Basic Event Cutsets for Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_{23}PQI	T_{23}*P*Q*VGA2*VGB2*RECOVERY	5.0×10^{-7}
	T_{23}*P*Q*VGB2*SSA*RECOVERY	3.4×10^{-7}
	T_{23}*P*Q*VGA2*SSB*RECOVERY	3.4×10^{-7}
	T_{23}*P*Q*VGA2*VGB1*RECOVERY	2.5×10^{-7}
	T_{23}*P*Q*VGA1*VGB2*RECOVERY	2.5×10^{-7}
	T_{23}*P*Q*VGA2*LB2*RECOVERY	2.3×10^{-7}
	T_{23}*P*Q*VGB2*LA2*RECOVERY	2.3×10^{-7}
	T_{23}*P*Q*SSA*SSB*RECOVERY	1.6×10^{-7}
	T_{23}*P*Q*VGB1*SSA*RECOVERY	1.3×10^{-7}
	T_{23}*P*Q*VGA1*SSB*RECOVERY	1.3×10^{-7}
	T_{23}*P*Q*LB2*SSA*RECOVERY	1.0×10^{-7}
	T_{23}*P*Q*LA2*SSB*RECOVERY	1.0×10^{-7}
	T_{23}*P*Q*VGA1*VGB1*RECOVERY	1.0×10^{-7}
	T_{23}*P*Q*VGA1*LB2*RECOVERY	8.5×10^{-8}
	T_{23}*P*Q*VGB1*LA2*RECOVERY	8.5×10^{-8}
	T_{23}*P*Q*SA*SB*RECOVERY	6.9×10^{-8}
	T_{23}*P*Q*LA2*LB2*RECOVERY	6.9×10^{-8}
	T_{23}*P*Q*VGA2*SBC*RECOVERY	3.2×10^{-8}
	T_{23}*P*Q*VGA2*BCACT*RECOVERY	3.2×10^{-8}
	T_{23}*P*Q*VGB2*SAC*RECOVERY	3.2×10^{-8}
	T_{23}*P*Q*VGB2*LRACT*RECOVERY	3.2×10^{-8}
	T_{23}*P*Q*SAC*SSB*RECOVERY	2.8×10^{-8}
	T_{23}*P*Q*SSB*LRACT*RECOVERY	2.8×10^{-8}
	T_{23}*P*Q*SBC*SSA*RECOVERY	2.8×10^{-8}
	T_{23}*P*Q*SSA*BCACT*RECOVERY	2.8×10^{-8}
	T_{23}*P*Q*PA27*VGB2*RECOVERY	2.2×10^{-8}
	T_{23}*P*Q*PB27*VGA2*RECOVERY	2.2×10^{-8}
	T_{23}*P*Q*VGA1*BCACT*RECOVERY	2.0×10^{-8}
	T_{23}*P*Q*VGA1*SBC*RECOVERY	2.0×10^{-8}
	T_{23}*P*Q*LRACT*VGB1*RECOVERY	2.0×10^{-8}
	T_{23}*P*Q*VGB1*SAC*RECOVERY	2.0×10^{-8}
	T_{23}*P*Q*PA27*SSB*RECOVERY	1.9×10^{-8}
	T_{23}*P*Q*PB27*SSA*RECOVERY	1.9×10^{-8}
	T_{23}*P*Q*SAACC*SB*RECOVERY	1.9×10^{-9}
	T_{23}*P*Q*LRACT*LB2*RECOVERY	1.9×10^{-8}
	T_{23}*P*Q*LA2*SBC*RECOVERY	1.9×10^{-8}
	T_{23}*P*Q*LB2*SAC*RECOVERY	1.9×10^{-8}
	T_{23}*P*Q*LA2*BCACT*RECOVERY	1.9×10^{-8}
	T_{23}*P*Q*SA*SBACC*RECOVERY	1.9×10^{-8}

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*Grand Gulf cut sets are taken directly from Grand Gulf RSSMAP study.

Table A11-3 (continued)

Grand Gulf 1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
SI	S*LA2*VGB2	2.8×10^{-7}
	S*SSA*SSB	1.9×10^{-7}
	S*VGB1*SSA	1.6×10^{-7}
	S*VGA1*SSB	1.6×10^{-7}
	S*LB2*SSA	1.3×10^{-7}
	S*LA2*SSB	1.3×10^{-7}
	S*VGA1*VGB1	1.3×10^{-7}
	S*VGA1*LB2	1.1×10^{-7}
	S*LA2*VGB1	1.1×10^{-7}
	S*SA*SB	8.6×10^{-8}
	S*LA2*LB2	8.6×10^{-8}
	S*VGA2*SBC	4.0×10^{-8}
	S*VGA2*BCACT	4.0×10^{-8}
	S*VGB2*SAC	4.0×10^{-8}
	S*VGB2*LRACT	4.0×10^{-8}
	S*SAC*SSB	3.5×10^{-8}
	S*LRACT*SSB	3.5×10^{-8}
	S*SSA*SBC	3.5×10^{-8}
	S*BCACT*SSA	3.5×10^{-8}
	S*PA27*VGB2	2.7×10^{-8}
	S*VGA2*PB27	2.7×10^{-8}
	S*VGA1*BCACT	2.5×10^{-8}
	S*VGA1*SBC	2.5×10^{-8}
	S*LRACT*VGB1	2.5×10^{-8}
	S*VGB1*SAC	2.5×10^{-8}
	S*PA27*SSB	2.4×10^{-8}
	S*PB27*SSA	2.4×10^{-8}
	S*SAACC*SB	2.4×10^{-8}
	S*LRACT*LB2	2.4×10^{-8}
	S*LA2*SBC	2.4×10^{-8}
	S*LB2*SAC	2.4×10^{-8}
	S*LA2*BCACT	2.4×10^{-8}
	S*SA*SBACC	2.4×10^{-8}
T_1QW		
	T_1*LOPNRE*LOPNRL*DIESEL1*DIESEL2* RECOVERY	1.1×10^{-6}
	T_1*LOPNRE*LOPNRL*SSA*DIESEL2*RECOVERY	6.4×10^{-7}
	T_1*LOPNRE*LOPNRL*DIESEL1*SSB*RECOVERY	6.4×10^{-7}
	T_1*LOPNRE*LOPNRL*VGB1*DIESEL1*RECOVERY	4.5×10^{-7}
	T_1*LOPNRE*LOPNRL*VGA1*DIESEL2*RECOVERY	4.5×10^{-7}
	T_1*LOPNRE*LOPNRL*SSA*SSB*RECOVERY	3.7×10^{-7}
	T_1*LOPNRE*LOPNRL*VGB1*SSA*RECOVERY	2.6×10^{-7}

Table A11-3 (continued)

Grand Gulf 1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_1QW (Cont.)	T_1*LOPNRE*LOPNRL*VGA1*SSB*RECOVERY	2.6×10^{-7}
	T_1*LOPNRE*LOPNRL*V1*DIESEL2*RECOVERY	2.4×10^{-7}
	T_1*LOPNRE*LOPNRL*DIESEL1*V2*RECOVERY	2.4×10^{-7}
	T_1*LOPNRE*LOPNRL*VGA1*VGB1*RECOVERY	1.9×10^{-7}
	T_1*LOPNRE*LOPNRL*SSA*V2*RECOVERY	1.4×10^{-7}
	T_1*LOPNRE*LOPNRL*V1*SSB*RECOVERY	1.4×10^{-7}
	T_1*LOPNRE*LOPNRL*VGB1*V1*RECOVERY	1.0×10^{-7}
	T_1*LOPNRE*LOPNRL*VGA1*V2*RECOVERY	1.0×10^{-7}
	T_1*LOPNRE*LOPNRL*V1*V2*RECOVERY	5.4×10^{-8}
	T_1*LOPNRE*LOPNRL*VGA2*DIESEL2*R *RECOVERY	3.7×10^{-8}
	T_1*LOPNRE*LOPNRL*VGB2*DIESEL1*R *RECOVERY	3.7×10^{-8}
	T_1*LOPNRE*LOPNRL*SAC*DIESEL2 *RECOVERY	3.6×10^{-8}
	T_1*LOPNRE*LOPNRL*DIESEL1*SBC *RECOVERY	3.6×10^{-8}
	T_1*LOPNRE*LOPNRL*BATB*DIESEL1 *RECOVERY	3.0×10^{-8}
	T_1*LOPNRE*LOPNRL*BATA*DIESEL2 *RECOVERY	3.0×10^{-8}
	T_1*LOPNRE*LOPNRL*VGA2*VGB2*R *RECOVERY	2.5×10^{-8}
	T_1*LOPNRE*LOPNRL*VGB2*SCVB*DIESEL1 *RECOVERY	2.3×10^{-8}
	T_1*LOPNRE*LOPNRL*VGA2*SCVA*DIESEL2 *RECOVERY	2.3×10^{-8}
T_2QW	T_{23}*Q*SSA*SSB*RECOVERY	3.2×10^{-6}
	T_{23}*Q*VGB1*SSA*RECOVERY	1.9×10^{-6}
	T_{23}*Q*VGA1*SSB*RECOVERY	1.9×10^{-6}
	T_{23}*Q*VGA1*VGB1*RECOVERY	9.4×10^{-7}
	T_{23}*Q*VGA2*VGB2*R*RECOVERY	3.0×10^{-7}
	T_{23}*Q*VGA2*SSB*R*RECOVERY	2.6×10^{-7}
	T_{23}*Q*VGB2*SSA*R*RECOVERY	2.6×10^{-7}
	T_{23}*Q*SSA*SBC*RECOVERY	2.6×10^{-7}
	T_{23}*Q*SAC*SSB*RECOVERY	2.6×10^{-7}
	T_{23}*Q*VGA2*VGB1*R*RECOVERY	1.9×10^{-7}
	T_{23}*Q*VGA1*VGB2*R*RECOVERY	1.9×10^{-7}
	T_{23}*Q*VGB1*SAC*RECOVERY	1.9×10^{-7}
	T_{23}*Q*VGA1*SBC*RECOVERY	1.9×10^{-7}
	T_{23}*Q*LA2*VGB2*R*RECOVERY	1.8×10^{-7}
	T_{23}*Q*VGA2*LB2*R*RECOVERY	1.8×10^{-7}
	T_{23}*Q*VGB2*SCVB*SSA*RECOVERY	1.7×10^{-7}
	T_{23}*Q*VGA2*SCVA*SSB*RECOVERY	1.7×10^{-7}

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Table A11-3 (continued)

Grand Gulf 1

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_2QW (Cont.)	T_{23}*Q*LA2*SSB*R*RECOVERY	1.5×10^{-7}
	T_{23}*Q*LB2*SSA*R*RECOVERY	1.5×10^{-7}
$T_{23}C$	T_{23}*C	5.4×10^{-6}

A12. Peach Bottom (WASH-1400 BWR) Dominant Sequences, System Level and Basic Event Cutsets, and Sequence Fault Trees.

Table A12-1

Peach Bottom (WASH 1400-BWR)

Description of Dominant Accident Sequences

Sequences	Description
TW	Failure of decay heat removal given transient event.
TC	Failure to make reactor subcritical given transient event.

Table A12-2

Peach Bottom

Systems Level Cutsets for Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
TW	T_3*LPIS*PCS	8.4E-6
	T_1*LPIS*PCS	4.8E-7
	T_3*HPSW*PCS	7.7E-6
	T_1*HPSW*PCS	4.4E-7
TC	T_3*RPS*OPERROR	1.3E-5

Table A12-3

Peach Bottom

Basic Event Cutsets for Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
TW	T*LPCI27*PCS	8.4E-6
	T*LPCI27*RLOP*PCSLOP	4.8E-7
	T*HPSW27*PCS	7.7E-6
	T*HPSW27*RLOP*PCSLOP	4.4E-7
TC	T*RPS*MSD	1.3E-5

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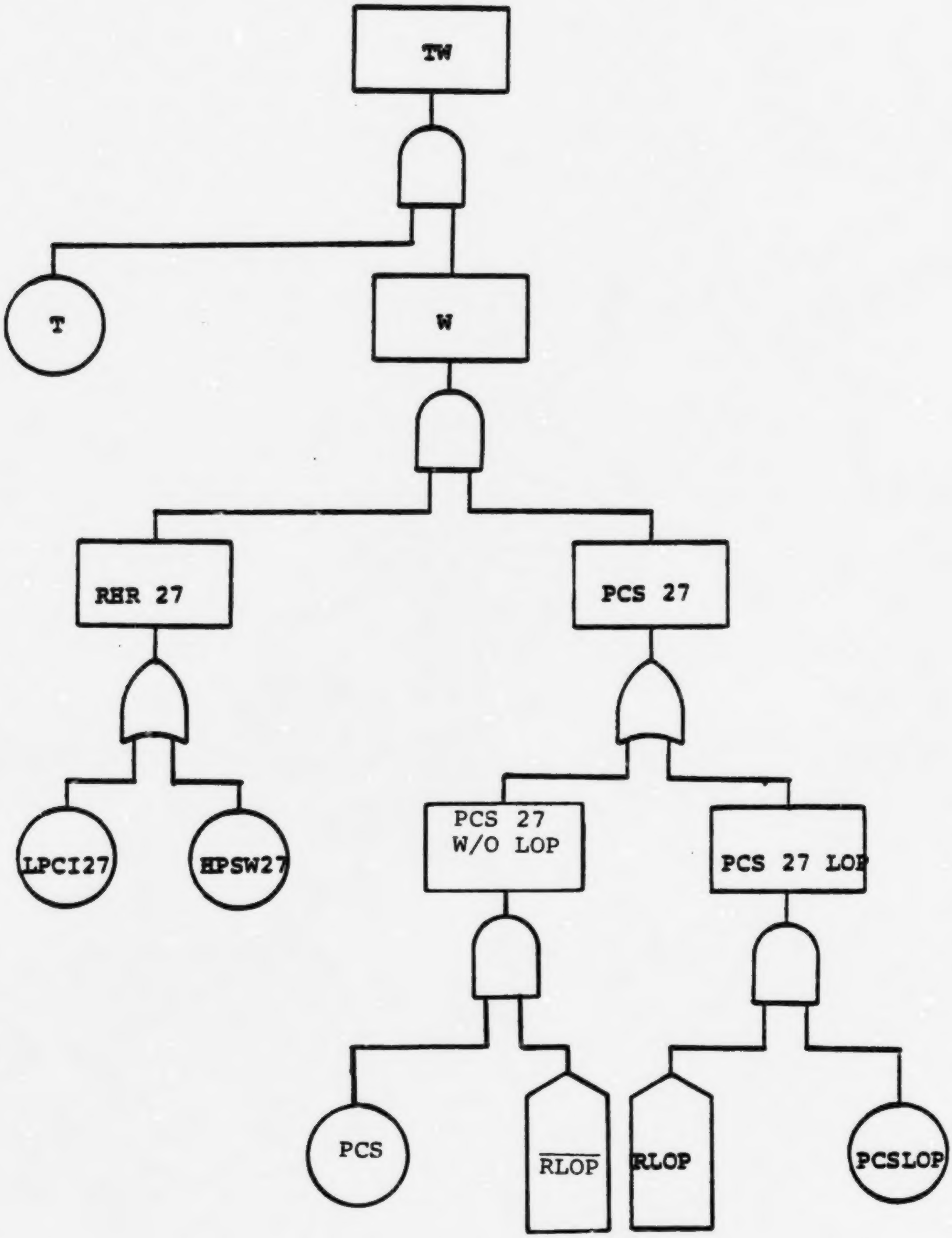
TW
T
W
RHR 27
PCS 27
LPCI27
HPSW27
PCS 27
W/O LOP
PCS 27 LOP
PCS
$\overline{\text{RLOP}}$
RLOP
PCSLOP

Figure A12-1. Sequence TW Fault Tree, Peach Bottom (WASH-1400 BWR)

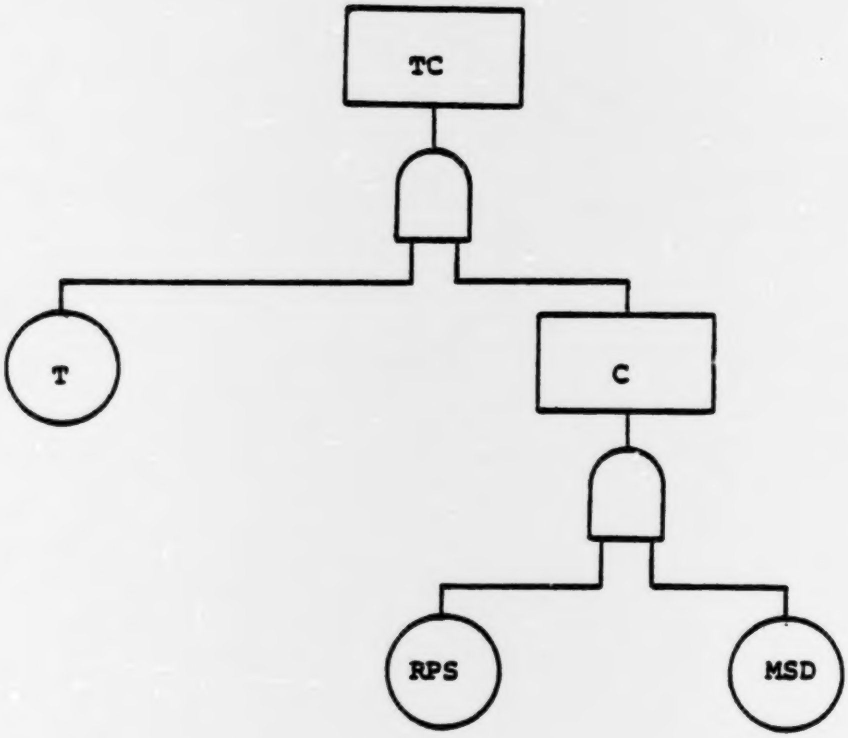
TC
T
C
RPS
MSD

Figure A12-2. Sequence to Fault Tree, Peach Bottom (WASH-1400 BWR)

A13. Limerick Dominant Sequences, System Level and Basic Event Cut Sets, and Sequence Fault Trees.

Table A13-1

Limerick Generating Station

Description of Dominant Accident Sequences

Sequences	Description
T_TQUX	Turbine trip transient followed by failure of Feedwater System (FWS), High Pressure Injection System (HPIS), and the Automatic Depressurization System (ADS).
T_FQUX	Transient involving MSIV closure, loss of FWS, or loss of Condenser, followed by failure of HPIS and ADS.
T_EUV	Loss of offsite power transients followed by failure of HPIS and LPIS.

Table A13-2

Limerick Generating Station

System Level Cut Sets for Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_TQUX	T_3*FWS*HPIS*ADS	7.7E-7
T_FQUX	T_2*FWS*HPIS*ADS	3.6E-6
T_EUV	T_1*HPIS*LPIS	5.9E-6

Table A13-3

Limerick Generating Station

Basic Events Cut Sets for Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_TQUX	TT*Q*U*X	7.7E-7
T_FQUX	TF*K*U*X	3.6E-6
T_EUV	TE*U*VV	5.9E-6

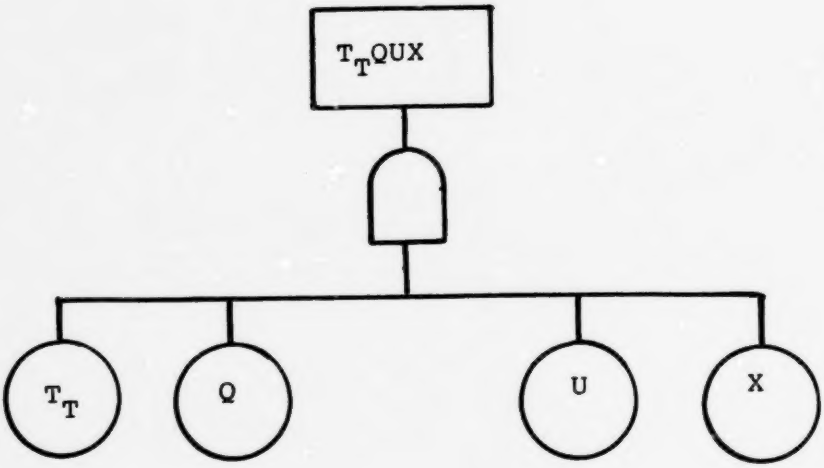
T_TQUX
T_T
Q
U
X

Figure A13-1. Sequence T_TQUX Fault Tree, Limerick

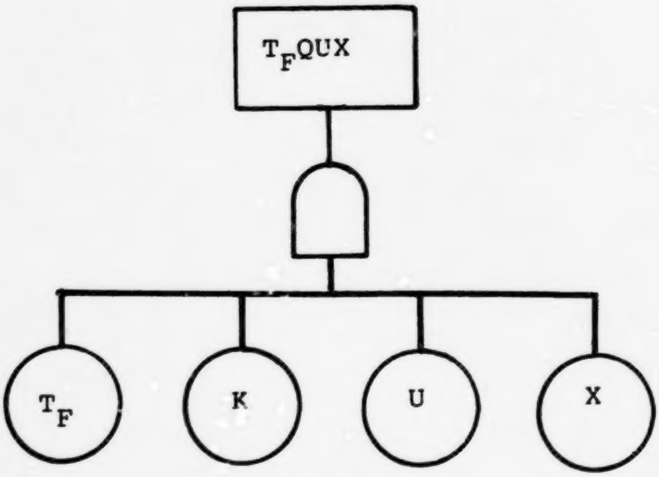
T_FQUX
T_F
K
U
X

Figure A13-2. Sequence T_FKUX Fault Tree, Limerick

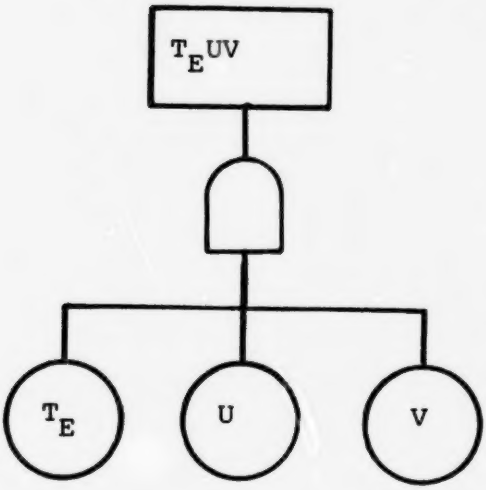

Figure A13-3. Sequence T_EUV Fault Tree, Limerick

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A14. Big Rock Point Dominant Sequences, System Level and Basic Event Cut Sets, and Sequence Fault Trees.

Table A14-1

Big Rock Point

Description of Dominant Accident Sequences

Sequences	Description
$I_L E_M L$	Interfacing systems LOCA in Recirculation System and Shutdown Cooling System, followed by failure of Emergency Condenser Make-up and the Post-Incident System (PIS).
$S_4 C$	Medium steam line break inside containment followed by failure of Reactor Depressurization System (RDS).
$D_P A$	Fire in Cable Penetration Area which effects all safety system cables.
$PE_m F_s KC$	Loss of offsite power followed by failure to restore power, failure of the RDS, and failure of Emergency Condenser.
$UE_m UKC$	Loss of Instrument Air, followed by failure of Emergency Condenser, RDS, and a safety valve fails to reseal.

Table A14-2

Big Rock Point

System Level Cut Sets for Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
I_LE_ML	V*PIS	2.0E-6
	V*PIS	8.1E-5
S_4C	S_1	1.0E-4
D_pA	FIRE	1.8E-4
PE_mF_sKC	T_1*LOPNRE*Q*OPERROR	7.0E-5
	T_1*LOPNRE*Q*OPERROR*FIRESYS	2.3E-5
UE_mUKC	U*OPERROR*RECOVERY*Q	8.6E-5
	U*FIRESYS*RECOVERY*Q	2.9E-5

Table A14-3

Big Rock Point

Basic Event Cut Sets For Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
I_1EML	I_L*PIS*ACLOCA	2.0E-6
	I_L*PIS10M*BCLOCA	8.1E-5
S_4C	S_4	1.0E-4
D_pA	Dp*A	1.8E-4
PE_mF_sKC	LOSP*FSEM*KEM*DMWP2B*OPVEC-1	7.0E-5
	LOSP*FSEM*KEM*DMWP2B*VEC-1	2.3E-5
UE_mUKC	U*OPVEC-1*U-NREM*KEM	8.6E-5
	U*VEC-1*U-NREM*KEM	2.9E-5

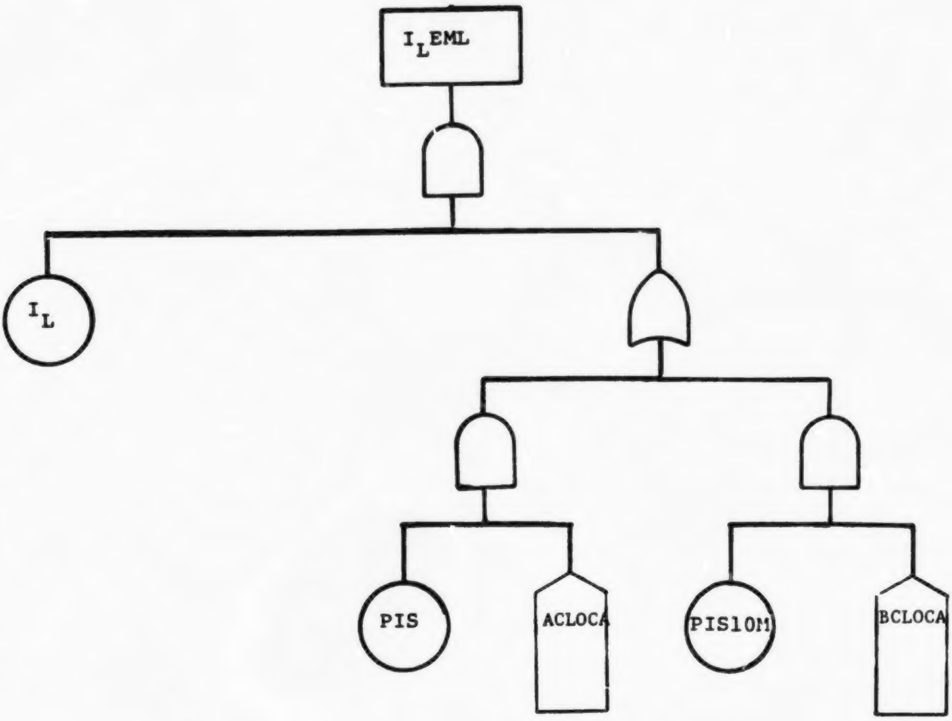
I_LEML
I_L
PIS
ACLOCA
PIS10M
BCLOCA

Figure A14-1. Sequence I_LEML Fault Trees, Big Rock Point

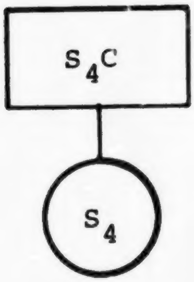

Figure A14-2. Sequence S_4C Fault Tree, Big Rock Point

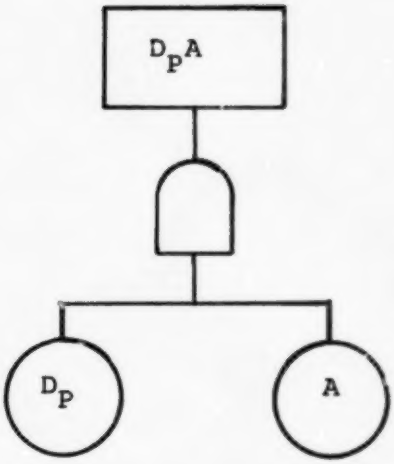

Figure A14-3. Sequence D_FA Fault Tree, Big Rock Point

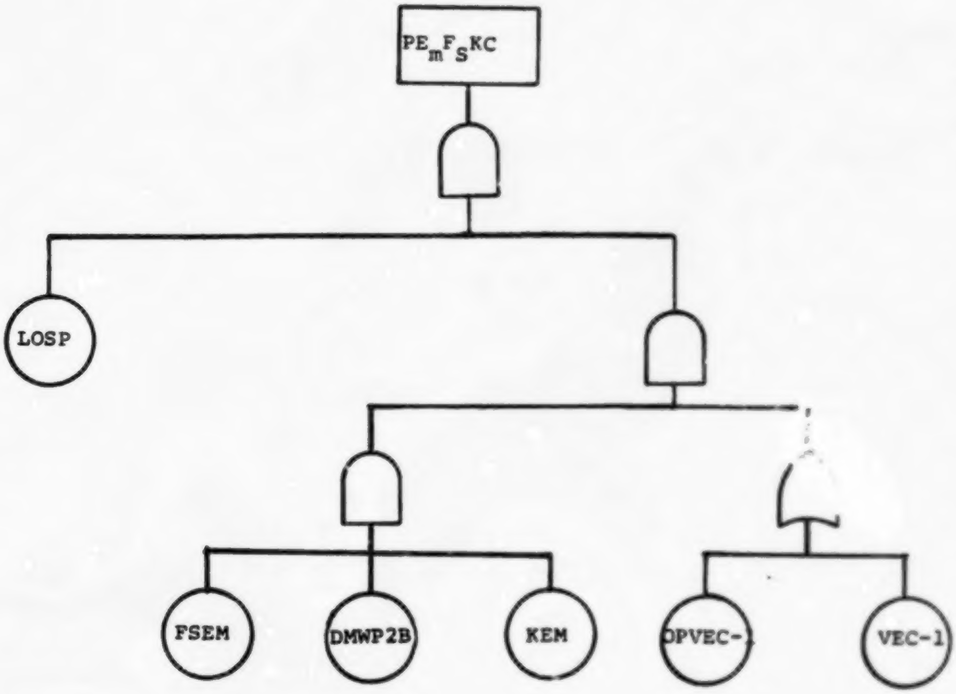
PE_mF_SKC
LOSP
FSEM
DMWP2B
KEM
DPVEC-
VEC-1

Figure A14-4. Sequence PE_mF_SKC Fault Tree, Big Rock Point

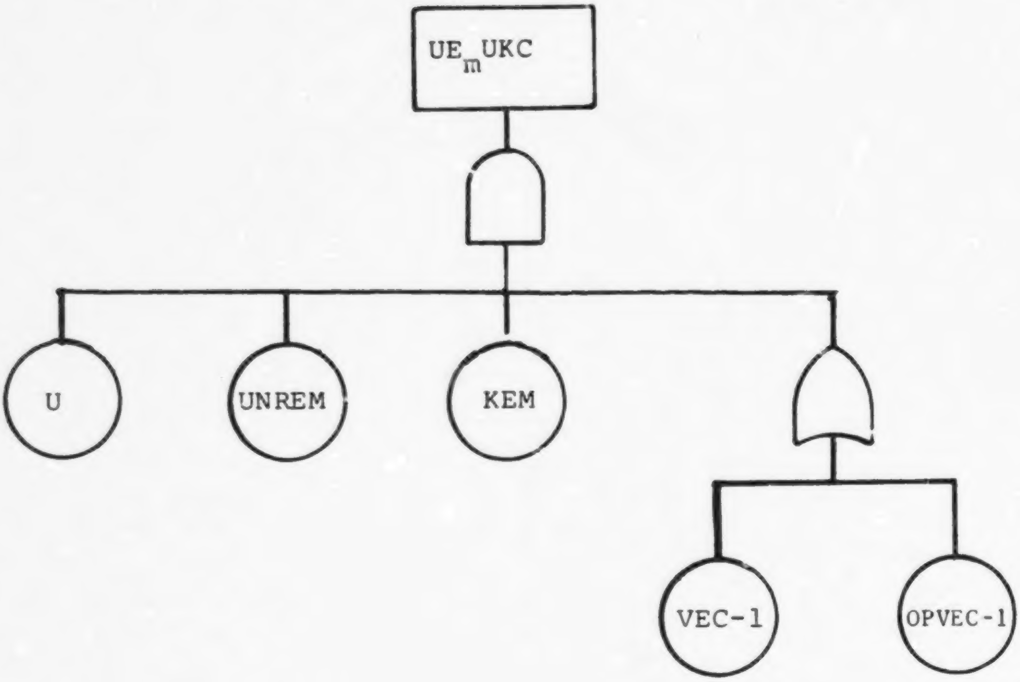
UE_mUKC
U
UNREM
KEM
VEC-1
OPVEC-1

Figure A14-5. Sequence UE_mUKC Fault Tree, Big Rock Point

A15. Millstone Dominant Sequences, System Level and Basic Event Cut Sets.

Table A15-1

Millstone

Description of Dominant Accident Sequences

Sequences	Description
T_4JCD	This sequence is initiated by loss of offsite power followed by failure of safety/relief valve to reseal, failure of Feedwater Coolant Injection System (FWCI) to provide coolant at high pressure, and operator failure to depressurize the reactor coolant system to allow low pressure coolant system to operate.
T_4JCEFG	Loss of offsite power followed by failure of safety/relief valve to reseal, failure of FWCI at high pressure, failure of Low Pressure Coolant Injection (LPCI) system and Cores Spray System (CS).
T_4KCD	Loss of offsite power followed by failure of Isolation Condensor (IC) to remove heat from reactor coolant system, failure of FWCI, and failure of Operator to depressurize reactor coolant system (RCS).
T_4KCEFG	Loss of offsite power followed by failure of IC to remove heat from RCS, failure of FWCI, failure of LPCI and CS at low pressure.
T_4LCD	Loss of offsite power, followed by failure of supply water to shell side of IC, failure of FWCI, operator failure to depressurize RCS.
T_2A	Loss of power conversion system followed by failure of reactor protection system (RPS).
T_4JCDG	Loss of offsite power followed by failure of FWCI, safety/relief valve to reseal, operator failure to depressuirze RCS, and failure of Containment Cooling System.
T_4JCMG	Loss of offsite power followed by FWCI failure, a safety/relief valve to reseal, failure of the shutdown cooling system to remove decay heat (LPCI), and failure of Containment Cooling System.

Table A15-1 (continued)

Millstone

Description of Dominant Accident Sequences

Sequences	Description
T_4LCMG	Loss of offsite power followed by failure of IC make-up system to supply water to the IC, and failure of FWCI. Later failure of LPCI and containment cooling system to remove decay heat occurs.
T_4KCMG	Loss of offsite power followed by failure of FWCI, failure of IC to remove heat from RCS, and failure of LPCI to remove decay heat, and failure of containment spray.

Table A15-2

Millstone

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_4JCD	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*EACPOWER	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*EACPOWER	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*EACPOWER	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*EACPOWER	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*FWCI	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*FWCI	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*FWCI	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*FWCI	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*EACPOWER	6.5E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*SWS	2.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*SWS	1.8E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*OPERROR*SWS	1.8E-6
T_4JCEFG	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	6.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER	3.1E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER	3.1E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	5.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	1.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	1.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*EACPOWER*SWS	1.6E-6
T_4KCD	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	2.4E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	8.2E-8
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	8.2E-8
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	8.2E-8
	T_4*.5HR-FTR-OS-PWR*OPERROR*IC*EACPOWER	8.2E-8

Table A15-2 (Cont.)

Millstone

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency $(R\text{-}yr)^{-1}$
T_4KCEFG	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	2.6E-6
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	1.3E-6
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	7.4E-7
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	7.4E-7
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	7.4E-7
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	3.7E-7
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	3.7E-7
	T_4*.5HR-FTR-OS-PWR*DCPOWER*EACPOWER	3.7E-7
	T_4*.5HR-FTR-OS-PWR*IC*EACPOWER*SWS	8.4E-7
	T_4*.5HR-FTR-OS-PWR*IC*EACPOWER*SWS	8.4E-7
T_4LCD	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.8E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.8E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.8E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.8E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*SWS	1.4E-6
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*SWS	9.7E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*SWS	9.7E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.7E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.7E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	9.7E-7
	T_4*2HR-FTR-OS-PWR*OPERROR*IC*FWCI	5.8E-7
T_2A	T_2*RPS	2.0E-5
T_4JCDG	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	1.3E-6
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	6.5E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	6.5E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	3.7E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	1.8E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	1.8E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	3.7E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	1.8E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	1.8E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER	3.7E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER*SWS	4.6E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER*SWS	1.3E-7
	T_4*.5HR-FTR-OS-PWR*OPERROR*S/R-VR*EACPOWER*SWS	1.3E-7

Table A15-2 (continued)

Millstone

System Level Cut Sets of Significant Sequences

Sequence	Cut Sets	Frequency (R-yr)
T_4JCMG	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	2.2E-6
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	1.1E-6
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	1.1E-6
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	6.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	3.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	3.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	6.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	3.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*EACPOWER	3.2E-6
T_4LCMG	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	1.2E-6
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	6.0E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	6.0E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	3.3E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	1.7E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	1.7E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	3.3E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	1.7E-7
	T_4*20HR-FTR-OS-PWR*EACPOWER*IC	1.8E-6
T_4KCMG	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	2.8E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	1.4E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	1.4E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	2.8E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	1.4E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	1.4E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	9.6E-8
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	4.8E-8
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	4.8E-8
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	2.8E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	1.4E-7
	T_4*20HR-FTR-OS-PWR*IC*EACPOWER	1.4E-7

Table A15-3

Millstone

Basic Event Cut Sets of Significant Sequences*

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_4JCD	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-1S14C-1-FTO	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-1S14C-2-FTO	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14CT-16-FTC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14AT2-1-FTC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14C-3-FRC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14C-4-FRC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14A-3-FRC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-E4-3C-FRC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-GTG-LOF	6.5E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*SW-448-STR-LOF	2.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*SW-9-MOV-FTC	1.8E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-EF7-26-FTC	1.8E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*MW-97-SOV-FRO	1.8E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-E3-5E-FRC	1.8E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14C-15-FRC	1.8E-6
T_4JCEFG	T_4*.5HR-FTR-OS-PWR*S/R-VR*AC-GTG-LOF*SW-DGN-HTX-LOF	6.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*92662-1A-XFO	3.1E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*92662-1-XFC	3.1E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*SW-DGN-HTX-LOF*AC-1S14C-1-FTO	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*SW-DGN-HTX-LOF*AC-1S14A-2-FTO	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*SW-DGN-HTX-LOF*AC-14CT-16-FTC	1.9E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*AC-GTG-LOF*AC-DGN*LOF	5.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*AC-DGN-LOF*AC1S14C-1-FTO	1.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*AC-DGN-LOF*AC1S14A-2-FTO	1.6E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*AC-DGN-LOF*AC14CT-16-FTC	1.6E-6
T_4KCD	T_4*.5HR-FTR-OS-PWR*MDP*IC-1-MOV-TMC*AC-14CT-16-FTC	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-1-MOV-TMC*AC-14AT2-1-FTC	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-1-MOV-TMC*AC-1S14A-2-FTO	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-1-MOV-TMC*AC-1S14C-1-FTO	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-2-MOV-TMC*AC-14CT-16-FTC	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-2-MOV-TMC*AC-14AT2-1-FTC	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-2-MOV-TMC*AC-1S14A-2-FTO	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-2-MOV-TMC*AC-1S14C-1-FTO	2.4E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-3-MOV-TMC*AC-14CT-16-FTC	8.2E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-3-MOV-TMC*AC-14AT2-1-FTC	8.2E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-3-MOV-TMC*AC-1S14A-2-FTO	8.2E-7
	T_4*.5HR-FTR-OS-PWR*MDP*IC-3-MOV-TMC*AC-1S14C-1-FTO	8.2E-7

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*Millstone cutsets are expanded from the "super sets" published in the PRA.

Table A15-3 (continued)

Millstone

Basic Event Cut Sets of Significant Sequences*

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_4KCEFG	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-TMO*AC-GTG-LOF	2.6E-6
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-LOF*AC-GTG-LOF	1.3E-6
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-TMO*AC-1S14C-1-FTO	7.4E-7
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-TMO*AC-1S14A-2-FTO	7.4E-7
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-TMO*AC-14CT-16-FTC	7.4E-7
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-LOF*AC-1S14C-1-FTO	3.7E-7
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-LOF*AC-1S14A-2-FTO	3.7E-7
	T_4*.5HR-FTR-OS-PWR*DC-A-BATT-LOF*AC-14CT-16-FTC	3.7E-7
	T_4*.5HR-FTR-OS-PWR*AC-GTG-LOF*IC-1-MOV-TMC*SW-DGN-HTX-LOF	8.4E-7
	T_4*.5HR-FTR-OS-PWR*AC-GTG-LOF*IC-2-MOV-TMC*SW-DGN-HTX-LOF	8.4E-7
T_4LCD	T_4*2HR-FTR-OS-PWR*MDP*AC-14C-3-FRC*ICM-10-MOV-FTO	9.8E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-14C-4-FRC*ICM-10-MOV-FTO	9.8E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-14A-3-FRC*ICM-10-MOV-FTO	9.8E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-E4-3C-FRC*ICM-10-MOV-FTO	9.8E-7
	T_4*2HR-FTR-OS-PWR*MDP*SW-448-STR-LOF*ICM-10-MOV-FTO	1.4E-6
	T_4*2HR-FTR-OS-PWR*MDP*SW-9-MOV-FTC*ICM-10-MOV-FTO	9.7E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-EF7-26-FTC*ICM-10-MOV-FTO	9.7E-7
	T_4*2HR-FTR-OS-PWR*MDP*MW-97-SOV-FRP*ICM-10-MOV-FTO	9.7E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-E3-5E-FRC*ICM-10-MOV-FTO	9.7E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-14C-15-FRC*ICM-10-MOV-FTO	9.7E-7
	T_4*2HR-FTR-OS-PWR*MDP*AC-GTG-LOF*FTR-ICM-10	2.1E-6
T_2A	T_2*A	2.0E-5
T_4JCDG	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-GTG-LOF*AC-IV-XFR-LOF	1.3E-6
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-GTG-LOF*AC-IV-3-FRC	6.5E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-GTG-LOF*AC-F5-1F-FRC	6.5E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-1S14C-1-FTO*AC-IV-XFR-LOF	3.7E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-1S14C-1-FTO*AC-IV-3-FRC	1.8E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-1S14C-1-FTO*AC-F5-1F-FRC	1.8E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-154A-2-FTO*AC-IV-XFR-LOF	3.7E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-154A-2-FTO*AC-IV-3-FRC	1.8E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-154A-2-FTO*AC-F5-1F-FRC	1.8E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14CT-16-FTC*AC-IV-XFR-LOF	3.7E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14CT-16-FTC*AC-IV-3-FRC	1.8E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-14CT-16-FTC*AC-F5-1F-FRC	1.8E-7
	T_4*.5HR-FTR-OS-PWR*S/R-VR*MDP*AC-GTG-LOF*AC-IAC-ABT-LOF	2.0E-6

Table A15-3 (continued)

Millstone

Basic Event Cut Sets of Significant Sequences*

Sequence	Cut Sets	Frequency (R-yr)$^{-1}$
T_4JCMG	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-GTG-LOF*AC-IV-XFR-LOF	2.2E-6
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-GTG-LOF*AC-IV-3-FRC	1.1E-6
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-GTG-LOF*AC-F5-1F-FRC	1.1E-6
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-1S14A-2-FTO*AC-IV-XFR-LOF	6.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-1S14A-2-FTO*AC-IV-3-FRC	3.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-1S14A-2-FTO*AC-F5-1F-FRC	3.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-1S14C-1-FTO*AC-IV-XFR-LOF	6.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-1S14C-1-FTO*AC-IV-3-FRC	3.1E-7
	T_4*20HR-FTR-OS-PWR*S/R-VR*AC-GTG-LOF*AC-IAC-ABT-LOF	3.2E-6
T_4LCMG	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*AC-IV-XFR-LOF*FTR-ICM-10	1.2E-6
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*AC-IV-3-FRC	6.0E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*AC-F5-1F-FRC	6.0E-7
	T_4*20HR-FTR-OS-PWR*AC-1S14A-2-FTO*AC-IV-XFR-LOF	3.3E-7
	T_4*20HR-FTR-OS-PWR*AC-1S14A-2-FTO*AC-IV-3-FRC	1.7E-7
	T_4*20HR-FTR-OS-PWR*AC-1S14A-2-FTO*AC-F5-1F-FRC	1.7E-7
	T_4*20HR-FTR-OS-PWR*AC-1S14A-2-FTO*AC-IV-XFR-LOF	3.3E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*AC-IAC-ABT-LOF	1.8E-6
T_4KCMG	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-1-MOV-TMC*AC-IV-XFR-LOF	2.8E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-1-MOV-TMC*AC-IV-3-FRC	1.4E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-1-MOV-TMC*AC-F5-1F-FRC	1.4E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-2-MOV-TMC*AC-IV-XFR-LOF	2.8E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-2-MOV-TMC*AC-IV-3-FRC	1.4E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-2-MOV-TMC*AC-F5-1F-FRC	1.4E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-3-MOV-TMC*AC-IV-XFR-LOF	9.8E-8
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-3-MOV-TMC*AC-IV-3-FRC	4.8E-8
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-3-MOV-TMC*AC-F5-1F-FRC	4.8E-8
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-4-MOV-TMC*AC-IV-XFR-LOF	2.8E-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-4-MOV-TMC*AC-IV-3-FRC	1.4D-7
	T_4*20HR-FTR-OS-PWR*AC-GTG-LOF*IC-4-MOV-TMC*AC-F5-1F-FRC	1.4E-7

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13. ABSTRACT *(200 words or less)*

This is a report on the project "Determination of Event Sequences Contributing to Overall Frequency of Core Melt." It is an analysis of 13 existing probabilistic risk assessment (PRAs) covering 15 power plants. The object of this project is to identify the plant systems and system components which contribute most to the overall core melt frequency and to some measure of risk. The results of this analysis are grouped into two major types of commercial reactors, boiling water reactors (BWRs), and pressurized water reactors (PWRs). Within each type of reactor, plants are grouped by manufacturer. The results are first analyzed on a plant-by-plant basis. Then, the results are discussed for all plants within each plant manufacturer type to study any interrelationships or common trends which might exist among plants built by the same manufacturer. The results are then discussed among all PWRs and all BWRs.

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